

## SEARCH REQUEST FORM

Scientific and Technical Information Center

Requester's Full Name: Harry Wilkins Examiner #: 78403 Date: 7/9/02  
Art Unit: 1742 Phone Number 305-9927 Serial Number: 09/773,782  
Mail Box and Bldg/Room Location: CP3-7326 Results Format Preferred (circle): PAPER DISK E-MAIL

If more than one search is submitted, please prioritize searches in order of need.

\*\*\*\*\*  
Please provide a detailed statement of the search topic, and describe as specifically as possible the subject matter to be searched. Include the elected species or structures, keywords, synonyms, acronyms, and registry numbers, and combine with the concept or utility of the invention. Define any terms that may have a special meaning. Give examples or relevant citations, authors, etc, if known. Please attach a copy of the cover sheet, pertinent claims, and abstract.

Title of Invention: Creep Resistant Zirconium Alloy and Nuclear Fuel Cladding Incorporating Same  
Inventors (please provide full names): Raymond Grant Rowe, Ronald Bert Adamson,  
Shaikh Tahir Mahmood  
Earliest Priority Filing Date: 02/02/2001

\*For Sequence Searches Only\* Please include all pertinent information (parent, child, divisional, or issued patent numbers) along with the appropriate serial number.

Attached claims 1-7

Also Zr-alloy can be Zircaloy-2 or ~~Zircaloy~~  
Zircaloy-4, Zr-2.5Nb, Reactor Grade  
Zirconium

## STAFF USE ONLY

Searcher: Colue  
Searcher Phone #: \_\_\_\_\_  
Searcher Location: \_\_\_\_\_  
Date Searcher Picked Up: \_\_\_\_\_  
Date Completed: 7/12/02  
Searcher Prep & Review Time: 3 hr  
Clerical Prep Time: \_\_\_\_\_  
Online Time: 4 hrs

## Type of Search

NA Sequence (#) \_\_\_\_\_  
AA Sequence (#) \_\_\_\_\_  
Structure (#) \_\_\_\_\_  
Bibliographic ☒ \_\_\_\_\_  
Litigation \_\_\_\_\_  
Fulltext \_\_\_\_\_  
Patent Family \_\_\_\_\_  
Other \_\_\_\_\_

## Vendors and cost where applicable

STN ☒ \_\_\_\_\_  
Dialog ☒ \_\_\_\_\_  
Questel/Orbit \_\_\_\_\_  
Dr.Link ☒ \_\_\_\_\_  
Lexis/Nexis \_\_\_\_\_  
Sequence Systems \_\_\_\_\_  
WWW/Internet \_\_\_\_\_  
Other (specify) \_\_\_\_\_

=> d his

(FILE 'HOME' ENTERED AT 10:47:43 ON 12 JUL 2002)

FILE 'REGISTRY' ENTERED AT 10:47:54 ON 12 JUL 2002

L1 55532 S ZR/ELS AND AYS/CI  
L2 53412 S ZR AND AYS/CI  
L3 45373 S L1 AND 50-100/MAC

FILE 'HCAPLUS' ENTERED AT 10:48:53 ON 12 JUL 2002

L4 1297849 S ALPHA?  
L5 59969 S L4(3N) (PHASE? OR STRUCTUR? OR MICROSTRUCTUR?)  
L6 178683 S MICROSTRUCTUR?  
L7 QUE PRODUC? OR PROD# OR GENERAT? OR MANUF? OR MFR# OR CREAT? OR  
L8 1084859 S CREEP? OR STRESS? OR STRAIN? OR DEFORMAT? OR FATIGUE? OR FRAC  
L9 176174 S L8(4N) (RESIST? OR RECOVER? OR STRENGTH?) OR TOUGHNESS? OR RES  
L10 737678 S NUCLEAR?  
L11 193290 S L10(4N) FUEL? OR URANIUM? OR PLUTONIUM OR PU  
L12 5494 S L10(3N) FUEL?(3N) CLAD?  
L13 QUE CLAD? OR CASING? OR SHEATH? OR ENSHEATH? OR ENCAS? OR ENCAP  
L14 695326 S LATH? OR STRIP? OR SWATH# OR BAND## OR SLAT### OR ROW###  
L15 694636 S TUBE# OR TUBING# OR TUBUL? OR TUBIFORM? OR PIPE# OR PIPING#  
L16 27688 S (FAST## OR SWIFT## OR RAPID? OR QUICK?) (4N) (COOL?)  
L17 25940 S (COLD# OR METAL?) (2N) (WORK? OR METALWORK?)  
L18 250121 S ANNEAL? OR RECRYSTALLI?  
L19 19467 S ZIRCALLOY? OR (ZIRONI## OR ZR) (3N) (ALLOY? OR AMALGAM? OR MIXTU  
L20 46961 S L3  
L21 57035 S L19 OR L20 OR ZIRCALLOY(2W)4  
L22 7517 S ACICULAR  
L23 4269 S NEEDLE?(3N) (LIKE#)  
L24 11713 S L22 OR L23  
L25 QUE HEAT? OR WARM? OR HOT# OR CALEFACT? OR TORREFACT? OR PYROL?

FILE 'STNGUIDE' ENTERED AT 11:05:39 ON 12 JUL 2002

FILE 'HCAPLUS' ENTERED AT 11:15:58 ON 12 JUL 2002

L26 946440 S 70/SC,SX OR 71/SC,SX  
L27 599536 S 56/SX,SC  
L28 QUE BETA  
L29 QUE (BINARY OR DUAL OR TWO) (3N) PHASE?  
L30 30738 S L21 AND L7  
L31 2577 S L30 AND L4  
L32 1144 S L30 AND L5  
L34 66 S L32 AND (L14 OR LATH?)  
L35 34 S L34 AND L8  
L36 7 S L34 AND L9  
L37 130 S L31 AND L14  
L38 56 S L37 AND L8  
L39 13 S L37 AND L9  
L40 1 S L39 AND L10  
L41 2802 S L21 AND L12  
L42 292 S L41 AND L4  
L43 120 S L41 AND L5  
L44 190 S L42 AND L7  
L45 71 S L43 AND L7  
L46 190 S L44 AND L4  
L47 71 S L45 AND L5  
L48 71 S L44 AND L5  
L49 71 S L45 AND L4  
L50 80 S L46 AND L8

L51 4 S L46 AND L9  
 L52 71 S L47 OR L48 OR L49  
 L53 38 S L52 AND L8  
 L54 4 S L52 AND L9  
 L55 293 S L41 AND L18  
 L56 54 S L55 AND L17  
 L57 2 S L56 AND L16  
 L58 60 S L55 AND L4  
 L59 39 S L58 AND L28  
 L60 31 S L59 AND L7  
 L61 1 S L60 AND L14  
 L62 2 S L60 AND L9  
 L63 15 S L36 OR L40 OR L51 OR L54 OR L57 OR L61 OR L62  
 L64 5 S L39 NOT L63

=> d cost

COST IN U.S. DOLLARS

	SINCE FILE ENTRY	TOTAL SESSION
CONNECT CHARGES	61.38	117.61
NETWORK CHARGES	1.86	4.74
SEARCH CHARGES	0.00	16.00
DISPLAY CHARGES	49.00	49.00
	-----	-----
	112.24	187.35
CAPLUS FEE (5%)	5.52	8.29
	-----	-----
FULL ESTIMATED COST	117.76	195.64

DISCOUNT AMOUNTS (FOR QUALIFYING ACCOUNTS)

CA SUBSCRIBER PRICE

SINCE FILE ENTRY	TOTAL SESSION
-12.39	-12.39

IN FILE 'HCAPLUS' AT 11:34:34 ON 12 JUL 2002

Harry,

I did a search in Chemical abstracts and another in Dialog.  
I also included a Derwent record I got on EAST. By searching  
(nuclear\$ near3 fuel\$ near3 clad\$) AND (zirconi\$3 or zircaloy\$3)  
I found a lot of good art (manuf. zircaloy cladding for nuclear  
applications).

I also included Derwent record so you look at Manual codes (CPI  
codes). K05-B04B and M26-B06C are derwent codes. The are derwents  
classification system. K05 is for nuclear reactors and M26 is for  
nonferrous alloys. To search with codes, just end with .cpi., for  
example M26-B06C.cpi. And of course you can truncate as well.

It is very powerful, and you can pull up good art very quickly.  
If you have any questions, feel free to call anytime.

John

? show file

File 6:NTIS 1964-2002/Jul W3  
(c) 2002 NTIS, Intl Cpyrght All Rights Res  
File 2:INSPEC 1969-2002/Jul W1  
(c) 2002 Institution of Electrical Engineers  
File 8:Ei Compendex(R) 1970-2002/Jul W1  
(c) 2002 Engineering Info. Inc.  
File 62:SPIN(R) 1975-2002/Jun W3  
(c) 2002 American Institute of Physics  
File 65:Inside Conferences 1993-2002/Jul W1  
(c) 2002 BLDSC all rts. reserv.  
File 77:Conference Papers Index 1973-2002/May  
(c) 2002 Cambridge Sci Abs  
File 94:JICST-EPlus 1985-2002/May W3  
(c)2002 Japan Science and Tech Corp(JST)  
File 103:Energy SciTec 1974-2002/Jun B2  
(c) 2002 Contains copyrighted material  
File 109:Nuclear Sci. Abs. 1948-1976  
(c)1997 Contains copyrighted material  
File 347:JAPIO Oct 1976-2002/Mar(Updated 020702)  
(c) 2002 JPO & JAPIO  
File 351:Derwent WPI 1963-2002/UD,UM &UP=200244  
(c) 2002 Thomson Derwent  
File 32:METADEX(R) 1966-2002/Aug B1  
(c) 2002 Cambridge Scientific Abs

? ds

Set	Items	Description
S1	3306	NUCLEAR?(3N)CLAD?
S2	6779	NUCLEAR?(3N)(CLAD? OR CASING? OR SHEATH? OR ENSHEATH? OR E-NCAS? OR ENCAPSULAT? OR ENVELOP? OR OVERLAID? OR LAMIN? OR LAMEL? OR ENCAS? OR WRAP? OR SURROUND?)
S3	1957570	NUCLEAR?
S4	65309	S3 AND (CLAD? OR CASING? OR SHEATH? OR ENSHEATH? OR ENCAS? OR ENCAPSULAT? OR ENVELOP? OR OVERLAID? OR LAMIN? OR LAMEL? OR ENCAS? OR WRAP? OR SURROUND?)
S5	87445	ZIRCALOY? OR (ZIRCONI? OR ZR) (4N) (ALLOY? OR AMALGAM?)
S6	63960	ALPHA(3N) (PHASE? OR STRUCTUR? OR MICROSTRUCTUR?)
S7	40457	S5 AND (PRODUC? OR PROD? ? OR GENERAT? OR MANUF? OR MFR? ?

OR CREAT? OR FORMING? ? OR FORMAT? OR MAKE? ? OR MADE? ? OR M-  
 AKING? ? OR FABRICAT? OR PREPAR? OR PREP? ?)

S8 3615733 TUBE? ? OR TUBING? ? OR TUBUL? OR TUBIFORM? OR TUBELIKE? OR  
 PIPE? ? OR PIPING? ? OR PIPELI? OR CONDUIT? OR CYLIND?

S9 1849880 LATH? OR STRIP? OR SWATH? OR BAND? OR SLAT?

S10 10038 ACIRCULAR? OR NEEDLE?(3N)LIKE?

S11 55548 (COARSE? OR LARGE? OR BIG?) (3N) (GRAIN?)

S12 174165 CREEP?

S13 16550 S12(4N) (RESIST?)

S14 3962 NUCLEAR?(5N)CLAD?

S15 24564 NUCLEAR? AND CLAD?

S16 122622 S3 AND S8

S17 7023 S7 AND S8

S18 2474 S17 AND CLAD?

S19 8723 S5 AND (PRODUC? OR PROD? ? OR GENERAT? OR MANUF? OR MFR? ?  
 OR CREAT? OR FORMING? ? OR FORMAT? OR MAKE? ? OR MADE? ? OR M-  
 AKING? ? OR FABRICAT? OR PREPAR? OR PREP? ?)/TI

S20 1625 S19 AND S3

S21 207 S19 AND S14

S22 5423 S5 AND (PRODUC? OR PROD? ? OR MANUF? OR MFR? ? OR FABRICAT?  
 OR PREPAR? OR PREP? ?)/TI

S23 176 S22 AND S14

S24 27 S23 AND ALPHA?

S25 137 S23 AND S8

S26 6 S25 AND S12

S27 2 S23 AND S9

S28 0 S25 AND LATH?

S29 0 S23 AND S10

S30 20 S23 AND GRAIN?

S31 3 S23 AND S11

S32 27 S23 AND COLD?(4N)WORK?

S33 8 S26 OR S31

S34 42 (S32 OR S24) NOT S33

S35 40 RD S34 (unique items)

? t s33/7,de/1-8

33/7,DE/1 (Item 1 from file: 6)  
 DIALOG(R)File 6:NTIS  
 (c) 2002 NTIS, Intl Cpyrght All Rights Res. All rts. reserv.

1191935 NTIS Accession Number: DE85011649  
 Manufacturing Process to Reduce Large Grain Growth in Zirconium Alloys  
 (Patent Application)  
 Rosecrans, P. M.  
 Department of Energy, Washington, DC.  
 Corp. Source Codes: 052661000  
 Report No.: PAT-APPL-6-636 659  
 Filed 1 Aug 84 11p  
 Languages: English Document Type: Patent  
 Journal Announcement: GRAI8521; NSA1000  
 This Government-owned invention available for U.S. licensing and,  
 possibly, for foreign licensing. Copy of application available NTIS.  
 Portions of this document are illegible in microfiche products.  
 NTIS Prices: PC A02/MF A01  
 Country of Publication: United States  
 Contract No.: AC12-76SN00052  
 It is an object of the present invention to provide a procedure for  
 desensitizing zirconium-based alloys to large grain growth (LGG) during

thermal treatment above the recrystallization temperature of the alloy. It is a further object of the present invention to provide a method for treating zirconium-based alloys which have been cold-worked in the range of 2 to 8% strain to reduce large grain growth. It is another object of the present invention to provide a method for fabricating a zirconium alloy clad nuclear fuel element wherein the zirconium clad is resistant to large grain growth. (ERA citation 10:030822)

Descriptors: \*Reactor Materials; \*Zirconium Base Alloys; Fabrication; Grain Growth; Heat Treatments; Metallurgy; Microstructure

33/7,DE/2 (Item 1 from file: 8)  
DIALOG(R)File 8:Ei Compendex(R)  
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05491054

E.I. No: EIP00035071486

Title: Influence of composition and fabrication process on out-of-pile and in-pile properties of M5 alloy

Author: Mardon, Jean-Paul; Charquet, Daniel; Senevat, Jean

Corporate Source: FRAMATOME Nuclear Fuel, Lyon, Fr

Conference Title: 12th ASTM International Symposium: Zirconium in the Nuclear Industry

Conference Location: Toronto, Que, Can Conference Date: 19980115-19980118

E.I. Conference No.: 56408

Source: ASTM Special Technical Publication n 1354 2000. p 505-524

Publication Year: 2000

CODEN: ASTTA8 ISSN: 1040-3094

Language: English

Document Type: JA; (Journal Article) Treatment: X; (Experimental)

Journal Announcement: 0004W3

Abstract: Within the scope of the optimization of the M5 cladding tubes made of ternary alloy (Zr, Nb, O), an extensive program of investigation and industrial development has been undertaken. The various possible factors and potential causes of variability have been thoroughly analyzed with the aid of industrial-scale or laboratory ingots from the point of view of their impact on the finished product properties. In this way, all the chemical composition variabilities of the alloying elements (Nb, O) and impurities (Fe, S, C) have been studied through variable-composition ingots. Also, a number of manufacturing process variants (number of melts, quench, extrusion, heat treatments, pilgering . . .) have been studied. In some cases, it was possible to investigate the combined effect of two types of parameters (sulfur-process and iron-process interactions). For each of the products manufactured in this way, systematic, characterization of: creep, microstructure (optical microscopy, TEM), corrosion (autoclave) tests was accomplished. In each case, the influence of each variability parameter was tested, and in many cases correlations with the out-of-pile characteristics of the finished tubes were established. Lastly, for some variables (process, S content, . . .) the effect of irradiation was more specifically analyzed. These investigations pointed to a new and very important factor, the effect of sulfur concentration on the in-pile operating properties, especially creep and growth. This set of results constitutes a database covering the whole industrial variability range of this alloy, allowing Framatome to embark upon its industrial development phase and to offer M5 cladding tube on the market. This product has been irradiated over a wide range of PWR service and environmental conditions in Europe and the U.S. The improvements in corrosion (Factors 3 to 4), hydriding (Factors 5 to 6), and creep and growth (Factors 2 to 3) data after five cycles (55 GWd/tU) show impressive gains over optimized low-tin

Zircaloy-4. (Author abstract) 12 Refs.

Descriptors: \*Zirconium alloys; Nuclear fuel cladding; Alloying elements; Niobium; Oxygen; Iron; Sulfur; Carbon; Crystal impurities; Ingots

33/7,DE/3 (Item 1 from file: 103)

DIALOG(R)File 103:Energy SciTec

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01738261 AIX-17-019614; EDB-86-061943

Author(s): Nakajima, Junjiro; Umehara, Hajime; Inagaki, Masatoshi

Title: Nuclear fuel cladding tubes and its fabrication

Corporate Source: Hitachi Ltd., Tokyo (Japan)

Patent No.: JP 60-36984 A

Patent Assignee(s): Hitachi Ltd., Tokyo, Japan

Patent Date Filed: Filed date 9 Aug 1983

Publication Date: 26 Feb 1985

p 6

Note: JP patent application 58-145860

Language: Japanese

Availability: JAPIO. Also available from INPADOC.

Abstract: The purpose of this patent is to improve the corrosion-resistance, stress corrosion-resistance and high temperature creep properties of fuel cladding tubes. In a fuel cladding tube having a fuel cladding layer made of a zirconium-based alloy, the outer surface layer, the inner surface layer and the intermediate layer between the outer and the inner surface layers is made of substantially complete recrystallization structure. The grain size is made larger in the order of the outer surface layer, intermediate layer and the inner surface layer.

Major Descriptors: \*FUEL CANS -- FABRICATION; \*FUEL CANS -- GRAIN SIZE

Descriptors: CORROSION RESISTANCE; CREEP; RECRYSTALLIZATION; STRESS CORROSION; ZIRCONIUM BASE ALLOYS

Broader Terms: ALLOYS; CHEMICAL REACTIONS; CORROSION; CRYSTAL STRUCTURE; MECHANICAL PROPERTIES; MICROSTRUCTURE; SIZE; ZIRCONIUM ALLOYS

33/7,DE/4 (Item 2 from file: 103)

DIALOG(R)File 103:Energy SciTec

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01587381 EDB-85-094160

Author(s): Rosecrans, P.M.

Title: Manufacturing process to reduce large grain growth in zirconium alloys

Corporate Source: General Electric Co., Schenectady, NY (USA)

Patent Assignee(s): Dept. of Energy

Application/Priority No.: US 6-636659

Publication Date: 1 Aug 1984

p 11

Order Number: DE85011649

Contract Number (DOE): AC12-76SN00052

Note: Portions of this document are illegible in microfiche products

Language: English

Availability: NTIS, PC A02/MF A01; 1.

Abstract: It is an object of the present invention to provide a procedure for desensitizing zirconium-based alloys to large grain growth (LGG) during thermal treatment above the recrystallization temperature of the alloy. It is a further object of the present invention to provide a method for treating zirconium-based alloys which have been cold-worked

in the range of 2 to 8% strain to reduce large grain growth. It is another object of the present invention to provide a method for fabricating a zirconium alloy clad nuclear fuel element wherein the zirconium clad is resistant to large grain growth.

Major Descriptors: \*REACTOR MATERIALS -- FABRICATION; \*ZIRCONIUM BASE ALLOYS -- FABRICATION; \*ZIRCONIUM BASE ALLOYS -- METALLURGY

Descriptors: GRAIN GROWTH; HEAT TREATMENTS; MICROSTRUCTURE

Broader Terms: ALLOYS; CRYSTAL STRUCTURE; MATERIALS; ZIRCONIUM ALLOYS

33/7,DE/5 (Item 1 from file: 347)  
DIALOG(R)File 347:JAPIO  
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04249278  
PRODUCTION METHOD FOR FUEL CLADDING

PUB. NO.: 05-240978 [JP 5240978 A] ,  
PUBLISHED: September 21, 1993 (19930921)  
INVENTOR(s): NAKAJIMA JUNJIRO  
UMEHARA HAJIME  
INAGAKI MASATOSHI  
APPLICANT(s): HITACHI LTD [000510] (A Japanese Company or Corporation), JP  
(Japan)  
APPL. NO.: 03-338154 [JP 91338154]  
FILED: December 20, 1991 (19911220)  
JAPIO CLASS: 23.1 (ATOMIC POWER -- General)

#### ABSTRACT

PURPOSE: To obtain a production method for fuel cladding for nuclear reactors superior in anti-corrosion, anti-stress corrosion and high temperature creep characteristic.

CONSTITUTION: To a pushed complex tube 22 consisting of a complex pilet assembled by a zirconium base alloy hollow pilet as an outer tube and a zirconium hollow pilet as an inner tube, sealed at both ends and treated in heat, a melt processing is applied with high frequency hardening using a high frequency induction heating coil 26 and a cooling nozzle 27 which move in the axial direction relatively to the outer surface of the pushed complex tube 22 in the state coolant is circulated inside. Then for repeating cold rolling and annealing by turns, melt processing is performed by supporting the upper part and the lower part of the pushed complex tube 22 with an upper support 23 and a lower support 24 with the same material as the complex pilet.

33/7,DE/6 (Item 1 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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012460096  
WPI Acc No: 1999-266204/199923  
Zirconium alloy nuclear fuel cladding production  
Patent Assignee: MITSUBISHI MATERIALS CORP (MITV )  
Inventor: ISOBE T; SUDA Y  
Number of Countries: 003 Number of Patents: 003  
Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
FR 2769637	A1	19990416	FR 9812784	A	19981013	199923 B
JP 11194189	A	19990721	JP 98287800	A	19981009	199939



US 6125161 A 20000926 US 98169968 A 19981013 200051  
US 99397094 A 19990916

Priority Applications (No Type Date): JP 98287800 A 19981009; JP 97278935 A 19971013

## Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
FR 2769637	A1		39	C21D-008/00	
JP 11194189	A		24	G21C-003/06	
US 6125161	A			G21C-003/07	Div ex application US 98169968

Abstract (Basic): FR 2769637 A1

## Abstract (Basic):

NOVELTY - In the production of nuclear fuel cladding of a zirconium alloy containing Nb or Nb+Ta, annealing is carried out at 550-850degreesC for 1-4 h such that the log of the cumulative anneal parameter is -20 to -15 and satisfies a mathematical relationship relating it to the Nb or Nb+Ta content.

DETAILED DESCRIPTION - Nuclear fuel cladding is produced by subjecting a zirconium alloy of composition (by wt.) 0.2-1.7% Sn, 0.18-0.6% Fe, 0.07-0.4% Cr, 0.05-1.0% Nb, optionally 0.01-0.1% Ta, balance zirconium and impurities, including at most 60 ppm N, to hot forging, solution heat treatment, hot extrusion, repeated annealing and cold rolling, and final stress relief annealing, the annealing being carried out at 550-850degreesC for 1-4 h such that the cumulative anneal parameter approximately  $SA_i$  (where  $SA_i = \log \left( \frac{t_i}{X_{Nb} - 0.2} \right)$ ) satisfies the relationships of  $SA_i = -20$  to  $-15$  and  $SA_i = -18 - 10X_{Nb}$  to  $-15 - 3.75(X_{Nb} - 0.2)$ , in which  $A_i$ =anneal parameter for the  $i$ 'th anneal,  $t_i$ =anneal duration (h) for the  $i$ 'th anneal,  $T_i$ =the anneal temperature (K) for the  $i$ 'th anneal and  $X_{Nb}$ =the Nb and optional Ta content (in wt.%). An INDEPENDENT CLAIM is also included for a zirconium alloy nuclear fuel cladding made by the above process.

USE - For producing nuclear fuel cladding tubes for a PWR.

ADVANTAGE - The annealing conditions provide the cladding tube with better corrosion resistance and creep properties than conventional cladding tubes and thus has a long useful life.

pp; 39 DwgNo 0/0

Title Terms: ZIRCONIUM; ALLOY; NUCLEAR; FUEL; CLAD; PRODUCE

Derwent Class: K05; M26; M29; X14

International Patent Class (Main): C21D-008/00; G21C-003/06; G21C-003/07

International Patent Class (Additional): C21D-001/26; C22C-016/00;

C22F-001/00; C22F-001/18

33/7,DE/7 (Item 2 from file: 351)

DIALOG(R)File 351:Derwent WPI

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010561475

WPI Acc No: 1996-058429/199606

Mfr. of of Zr alloys tubes for nuclear reactor fuel - by extruding, cold rolling and quenching achieving low irradiation induced axial growth with high transversal creep strength and good corrosion resistance during irradiation

Patent Assignee: SANDVIK AB (SANV )

Inventor: ANDERSSON T; ANDERSON T

Number of Countries: 019 Number of Patents: 010

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
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WO 9535395	A1	19951228	WO 95SE749	A	19950620	199606	B
SE 9402250	A	19960129	SE 942250	A	19940622	199615	
EP 760017	A1	19970305	EP 95923640	A	19950620	199714	
			WO 95SE749	A	19950620		
JP 10501846	W	19980217	WO 95SE749	A	19950620	199817	
			JP 96502068	A	19950620		
KR 97704064	A	19970809	WO 95SE749	A	19950620	199836	
			KR 96707338	A	19961221		
US 5876524	A	19990302	WO 95SE749	A	19950620	199916	
			US 97765590	A	19970417		
EP 760017	B1	19990908	EP 95923640	A	19950620	199941	
			WO 95SE749	A	19950620		
DE 69512052	E	19991014	DE 612052	A	19950620	199949	
			EP 95923640	A	19950620		
			WO 95SE749	A	19950620		
ES 2135749	T3	19991101	EP 95923640	A	19950620	199953	
SE 513488	C2	20000918	SE 942250	A	19940622	200054	

Priority Applications (No Type Date): SE 942250 A 19940622

Cited Patents: 03Jnl.Ref; EP 488027; US 2894866; US 4450020; US 5266131; WO 9501639

#### Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
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WO 9535395	A1	E	18	C22C-016/00	
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Designated States (National): JP KR US

Designated States (Regional): AT BE CH DE DK ES FR GB GR IE IT LU MC NL PT SE

SE 9402250	A			C22F-001/18	
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EP 760017	A1	E		C22C-016/00	Based on patent WO 9535395
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Designated States (Regional): DE ES FR GB IT SE

JP 10501846	W		15	C22F-001/18	Based on patent WO 9535395
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KR 97704064	A			C22C-016/00	Based on patent WO 9535395
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US 5876524	A			C22F-001/18	Based on patent WO 9535395
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EP 760017	B1	E		C22C-016/00	Based on patent WO 9535395
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Designated States (Regional): DE ES FR GB IT SE

DE 69512052	E			C22C-016/00	Based on patent EP 760017
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Based on patent WO 9535395

ES 2135749	T3			C22C-016/00	Based on patent EP 760017
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SE 513488	C2			C22F-001/18	
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#### Abstract (Basic): WO 9535395 A

Mfr. of cladding tubes and structural tube parts for fuel of Zr alloys for application in nuclear reactors is claimed. The Zr alloy is initially extruded and then followed by a series of cold rolling passes accompanied with alpha-annealing steps. beta-quenching is then performed by heating in the beta-phase range (950-1250 deg. C) until the structure is 100% beta-phase, then quenching at 100-450 deg. C s<sup>-1</sup>, transforming the entire tube to alpha-phase followed by a vacuum anneal in the alpha-phase for suitable time and temperature to produce an annealing parameter of 3.4x10<sup>-16</sup>x10<sup>-13</sup>.

USE - The method of manufacture relates to the production of cladding for nuclear core materials and structural parts in fuel element skeletons, using Zr based alloys.

ADVANTAGE - Compared to prior art material, this invention provides lower irradiation induced axial growth with higher transversal creep strength and good corrosion resistance during irradiation. Tests for corrosion resistance conducted in steam at 400 deg. C for 60 days the invention showed a weight gain of 2-4 mg dm<sup>-3</sup> less than prior art. Creep tests at 400 deg. C for 240 hours with a peripheral tension of 130 MPa. Transversal creep elongation of 0.45-0.70% was found in this

invention compared to 1.8-2.0% in the prior art. Axial growth was compared by the use of Kearns factor, fa. Axial growth is found to be greater in material with a lower fa factor. The invention has a fa value in the range 0.26-0.32 while conventional tubes have 0.03-0.07.

Dwg.0/0

Title Terms: MANUFACTURE; ALLOY; TUBE; NUCLEAR; REACTOR; FUEL; EXTRUDE;  
COLD; ROLL; QUENCH; ACHIEVE; LOW; IRRADIATE; INDUCE; AXIS; GROWTH; HIGH;  
TRANSVERSE; CREEP; STRENGTH; CORROSION; RESISTANCE; IRRADIATE  
Derwent Class: K05; M26; M29; X14  
International Patent Class (Main): C22C-016/00; C22F-001/18  
International Patent Class (Additional): C22F-001/00; G21C-003/06

33/7,DE/8 (Item 3 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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007736305

WPI Acc No: 1989-001417/198901

Zirconium alloy nuclear fuel cladding mfr. - involving final beta phase heat treatment for improved corrosion and creep resistance

Patent Assignee: COMMISSARIAT ENERGIE ATOMIQUE (COMS ); FRAMATOME SA (FRAT );  
URANIUM PECHINEY & FRAMATOME ZIRCOTUBE (UGIN ); FRAMATOME (FRAT );  
SNC URANIUM PECH & FRAMA (UGIN ); SNC URANIUM PECH & FRAMAT (UGIN )

Inventor: DECOURS J; MARDON J P; PELCHAT J; WEISZ M; LEPAPE J; LE PAPE J

Number of Countries: 011 Number of Patents: 008

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
EP 296972	A	19881228	EP 88401573	A	19880622	198901 B
JP 1097897	A	19890417	JP 88153689	A	19880623	198921
CN 1030261	A	19890111				198949
ZA 8804447	A	19900228	ZA 884447	A	19880622	199013
US 4938921	A	19900703	US 88210444	A	19880623	199029
EP 296972	B1	19920812	EP 88401573	A	19880622	199233
DE 3873643	G	19920917	DE 3873643	A	19880622	199239
			EP 88401573	A	19880622	
ES 2034312	T3	19930401	EP 88401573	A	19880622	199323

Priority Applications (No Type Date): FR 878814 A 19870623

Cited Patents: DE 2008320; DE 2651870; DE 2951096; DE 2951102; EP 213771;  
EP 98996; FR 2368547; US 4238251; US 423851

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
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EP 296972	A	F	8		
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Designated States (Regional): BE DE ES FR GB IT SE

EP 296972	B1	F	C22F-001/18		
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Designated States (Regional): BE DE ES FR GB IT SE

DE 3873643	G		C22F-001/18	Based on patent EP 296972	
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ES 2034312	T3		C22F-001/18	Based on patent EP 296972	
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Abstract (Basic): EP 296972 A

In the mfr. of zirconium alloy tubing for fuel cladding, using several successive rolling passes and anneals, the novelty is that the final stage is a beta phase homogenisation heat treatment at 950-1250 deg.C, followed by rapid cooling to ambient temp.

USE/ADVANTAGE - The process is used in the mfr. of PWR nuclear fuel cladding, has improved general corrosion resistance under irradiation at its external surface, improved internal corrosion resistance under stress and irradiation, improved radial creep resistance at high temps. and under neutron flux, improved axial creep resistance and reduced axial extension. The fuel elements can remain longer in the reactor

core since the cladding remains sealed and has long term resistance to irradiation.

0/0

Abstract (Equivalent): EP 296972 B

In the mfr. of zirconium alloy tubing for fuel cladding, using several successive rolling passes and anneals, the novelty is that the final stage is a beta phase homogenisation heat treatment at 950-1250 deg.C followed by rapid cooling to ambient temp.

USE/ADVANTAGE - The process is used in the mfr. of PWR nuclear fuel cladding, has improved general corrosion resistance under irradiation at its external surface, improved internal corrosion-resistance under stress and irradiation, improved radial creep resistance at high temps. and under neutron flux, improved axial creep resistance and reduced axial extension. The fuel elements can remain longer in the reactor core since the cladding remains sealed and has long term resistance to irradiation.a

Abstract (Equivalent): US 4938921 A

Zr alloy tube for a fuel element sheath in a nuclear reactor is mfd. from Zircalloy-4 alloy contg. (%) 1.2-1.7Sn, 0.18-0.24 Fe, 0.7-0.13 Cr, such that Fe-Cr = 0.28 min., with 80-270 ppm C and 900-1600 ppm O<sub>2</sub>. The tube is produced in a succession of cold rolling and annealing steps, including a final beta phase heat treatment in which the tube is maintained at 950-1250deg.C for a time sufficient to obtain a homogeneous beta phase through the wall thickness, followed by rapid cooling to ambient temp. to retain the beta phase.

ADVANTAGE - Enhanced corrosion resistance and creep resistance.

Title Terms: ZIRCONIUM; ALLOY; NUCLEAR; FUEL; CLAD; MANUFACTURE; FINAL; BETA; PHASE; HEAT; TREAT; IMPROVE; CORROSION; CREEP; RESISTANCE

Index Terms/Additional Words: ZIRCONIUM

Derwent Class: K05; M26

International Patent Class (Main): C22F-001/18

International Patent Class (Additional): C22C-016/00; G21C-003/06

? t s35/7,de/1-40

35/7,DE/1 (Item 1 from file: 2)

DIALOG(R)File 2:INSPEC

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5069574 INSPEC Abstract Number: A9521-8160B-035

Title: Remote Raman spectroscopic studies of corrosion products formed on nuclear fuel claddings used in PWR and AGR systems

Author(s): Edwards, H.G.M.; Long, D.A.; Willis, I.T.

Author Affiliation: Dept. of Chem. & Chem. Technol., Bradford Univ., UK

Journal: Journal of Raman Spectroscopy vol.26, no.8-9 p.757-62

Publication Date: Aug.-Sept. 1995 Country of Publication: UK

CODEN: JRSPAF ISSN: 0377-0486

U.S. Copyright Clearance Center Code: 0377-0486/95/080757-06

Language: English Document Type: Journal Paper (JP)

Treatment: Experimental (X)

Abstract: The construction of a remote Raman microscope system with a variable microscope-to-spectrometer distance of up to 3 m is described. Oxidative surface corrosion products on zirconium alloy and stainless-steel nuclear fuel claddings for pressurized water and advanced gas cooled reactor systems produced under various conditions were studied and the potential of the technique for in situ analysis is discussed. Under the conditions studied experimentally, the major corrosion products are found to be alpha -ZrO/sub 2/ and alpha -Fe/sub 2/O/sub 3/ for the zirconium alloy and stainless-steel claddings, respectively. (29 Refs)

Subfile: A

Descriptors: corrosion; fission reactor fuel claddings; oxidation; Raman spectra; stainless steel; zirconium alloys  
Copyright 1995, IEE

35/7,DE/2 (Item 1 from file: 8)  
DIALOG(R)File 8:EI Compendex(R)  
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05491051

E.I. No: EIP00035071483

Title: New fabrication process for Zr-lined Zircaloy-2 tubing

Author: Abe, Hideaki; Takeda, Kiyoko; Uehira, Akihiro; Anada, Hiroyuki;  
Furugen, Munekatsu

Corporate Source: Sumitomo Metal Industries Ltd, Hyogo, Jpn

Conference Title: 12th ASTM International Symposium: Zirconium in the Nuclear Industry

Conference Location: Toronto, Que, Can Conference Date:  
19980115-19980118

E.I. Conference No.: 56408

Source: ASTM Special Technical Publication n 1354 2000. p 425-459

Publication Year: 2000

CODEN: ASTTA8 ISSN: 1040-3094

Language: English

Document Type: JA; (Journal Article) Treatment: T; (Theoretical)

Journal Announcement: 0004W3

Abstract: A new fabrication process for Zr-lined Zircaloy-2 cladding tubes was developed, including a cold pilgering pass schedule and an appropriate heat treatment. In this study, the effect of tool design in cold pilgering on the quality of tubes was investigated. Cold pilgering tests were performed using tools with different tool curves. Simultaneously, for each tool the plastic strain and stress in the tubes during cold pilgering were simulated using a theoretical plastic deformation model. The investigations indicated that an abrupt change in the strain and stress in the tubes during cold pilgering should be avoided to prevent crack formation in the tubes. The results made possible the use of higher reduction (91%) in cold pilgering. Also, an appropriate heat treatment for the properties of the final tubes in the new process was investigated. It indicated that quenching the tubeshell and the intermediate annealing temperature had a small effect on the mechanical properties. To obtain both nodular and uniform corrosion resistance the appropriate precipitate size was in the range 140 to 170 nm. beta -quenching of the tubeshell was not effective for improving the uniform corrosion resistance. An appropriate intermediate annealing temperature was in the range 823 to 953 K for the alpha plus beta quenched and for the alpha -annealed tubeshells. These results led to the new fabrication process for Zr-lined Zircaloy-2 cladding tubes (12 mm outside diameter) in which the final tube was fabricated using only two cold pilgering passes from the tubeshell (63.5 mm outside diameter). (Author abstract) 19 Refs.

Descriptors: \*Zirconium alloys; Nuclear fuel cladding; Tubes (components); Cold rolling; Strain; Stress analysis; Plastic deformation; Quenching; Annealing; Corrosion resistance

35/7,DE/3 (Item 2 from file: 8)  
DIALOG(R)File 8:EI Compendex(R)  
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04213546

E.I. No: EIP95072798287

Title: Influence of manufacturing process on the in-reactor creep anisotropy of stress-relieved Zircaloy-2 cladding

Author: Shann, S.H.; Van Swam, L.F.

Corporate Source: Siemens Power Corp, Richland, WA, USA

Source: Nuclear Engineering and Design v 156 n 3 Jun 2 1995. p 351-358

Publication Year: 1995

CODEN: NEDEAU ISSN: 0029-5493

Language: English

Document Type: JA; (Journal Article) Treatment: T; (Theoretical); X; (Experimental)

Journal Announcement: 9509W4

Abstract: A procedure to determine the axial/radial and circumferential/radial contractile strain ratios (the R and P factors respectively in the Backofen-modified von Mises-Hill yield criterion) from post-irradiation dimensional measurements of Zircaloy-2 cladding of BWR fuel rods, tie rods and water rods was developed and has been described previously (S.H. Shann and L.F. van Swam, Creep anisotropy of Zircaloy-2 cladding during irradiation, Trans. SMiRT-11, Vol. C, 1991). The present study employs the procedure to determine the anisotropy factors R and P for textured cold-worked stress-relieved (CWSR) Zircaloy-2 cladding fabricated by various manufacturing processes. The analysis indicates that the cladding manufacturing process can have a pronounced effect on the anisotropy of irradiation-induced creep. Cladding types with identical yield and ultimate tensile strengths but fabricated by different manufacturing processes have different values of R and P during in-reactor creep. (Author abstract) 4 Refs.

Descriptors: \*Nuclear fuel cladding; Anisotropy; Zirconium alloys; Irradiation; Stresses; Processing; Creep; Strain; Mathematical models; Tensile strength

35/7,DE/4 (Item 3 from file: 8)

DIALOG(R)File 8:Ei Compendex(R)

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00189643

E.I. Monthly No: EI71X177920

Title: Effects of fabrication conditions on mechanical properties of zircaloy tubes for nuclear fuel cladding.

Author: KONISHI, T.; MATSUDA, K.

Source: Sumitomo Metals v 23 n 1 Jan 1971 p 40-8

Publication Year: 1971

CODEN: SUMMA

Language: JAPANESE

Journal Announcement: 71X1

Abstract: Mechanical properties and related features of zircaloy tubes are positively adjustable in wide range by controlling the fabrication conditions. In general, choice of the reducing process which would give large final reduction, large final Q value and large total Q value in cold working stage will yield versatile material that can meet varieties of specifications by adjusting subsequent finishing processes. RT value will be newly defined in the report. This value (analogous to r value used in deep drawing of sheet) is considered to be an effective measure in discussing the mechanical properties and related features of zircaloy tubes. In Japanese with English synopsis.

Descriptors: \*TUBES--\*Zircaloy; TUBES MANUFACTURE

35/7,DE/5 (Item 1 from file: 94)

DIALOG(R)File 94:JICST-EPlus

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00380277 JICST ACCESSION NUMBER: 87A0107292 FILE SEGMENT: JICST-E  
Manufacture and inspection techniques of Zircaloy nuclear fuel cladding  
tubes, and product quality.

TAKAISHI KAZUhide (1); MIYAJI MASATOSHI (1); WAKAMATSU RYUJI (1)

(1) Kobeseikoshō Chofukitakōjō

R & D / Kobe Seiko Giho(Kobe Steel Engineering Reports), 1987, VOL.37,NO.1  
, PAGE.5-9, FIG.9, TBL.2

JOURNAL NUMBER: F0164ABB ISSN NO: 0373-8868

UNIVERSAL DECIMAL CLASSIFICATION: 621.771.2/.8 621.039.54

LANGUAGE: Japanese

COUNTRY OF PUBLICATION: Japan

DOCUMENT TYPE: Journal

ARTICLE TYPE: Commentary

MEDIA TYPE: Printed Publication

ABSTRACT: As the result of extensive studies on fabrication techniques of  
nuclear fuel cladding tubes since 1957, the comprehensive worldwide  
supply of high-quality Zircaloy-2 fuel cladding tubes for BWRs has been  
established and annual production capacity has now reached  
650000meters. Recently, a fabrication technique of zirconium-lined  
Zircaloy-2 nuclear fuel cladding tubes has been developed, and  
qualification by customers was finished in 1986.(author abst.)

DESCRIPTORS: light water reactor; fuel can; mass production; tube rolling;  
cold rolling; degreasing; annealing; polishing(machining); acid  
cleaning; hardening(heat treatment); Zircaloy; zirconium; clad material  
; lining; corrosion resistance; local corrosion; blast cleaning; sand  
particle

BROADER DESCRIPTORS: thermal neutron reactor; nuclear reactor; fuel element  
; reactor component; production method; method; rolling(plastic  
working); plastic working; working and processing; tubemaking;  
manufacturing; cold working; removal; heat treatment; treatment;  
machining; chemical cleaning; cleaning(washing); cleaning(purification)  
; zirconium base alloy; nonferrous alloy; alloy; metallic material; 4A  
group element; transition metal; metallic element; element; material;  
operation(processing); resistance(endure); corrosion; sand; clastic  
sediment; sediment; soil; particle

35/7,DE/6 (Item 1 from file: 103)

DIALOG(R)File 103:Energy SciTec

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04163735 JPN-97-004434; EDB-97-072439

Title: Nuclear fuel cladding tube and method of manufacturing the same

Author(s)/Editor(s): Higashinakagawa, Emiko; Kubo, Hiroshi; Obata, Minoru;  
Hisatsune, Yoshimi.

Corporate Source: Toshiba Corp., Kawasaki, Kanagawa (Japan)

Patent No.: JP 9-5485 A

Patent Assignee(s): Toshiba Corp., Kawasaki, Kanagawa (Japan)

Priority No.: JP 7-155662

Patent Date Filed: 22 Jun 1995

Publication Date: 10 Jan 1997

(6 p)

Language: Japanese

Availability: Available from JAPIO. Also available from EPO

Abstract: A cladding tube main body made of a zirconium alloy and an end  
plug are joined by welding. Tensile stresses at the weld heat-affected  
portion between the cladding tube main body and the end plug are  
removed, so that compression stresses of 0 MPa or more but less than

the endurance strength of the zirconium alloy is applied on the weld heat affected portion. As the zirconium alloy, a zircaloy-2 or zircaloy-4 is preferable since it is excellent in the corrosion resistance and strength. The zirconium alloy may preferably be used also to the material of the end plug. The treatment for the removal of the tensile stresses includes a method of applying annealing to the weld heat-affected portion or a method of applying compression stresses thereto by applying external force such as a shot peening treatment. This can suppress occurrence of nodular corrosion and white homogeneous corrosion caused in the vicinity of the welded portion. (I.N.)

Major Descriptors: \*FUEL CANS -- WELDING

Descriptors: ANNEALING; CLOSURES; COMPRESSION; HEAT AFFECTED ZONE; SHOT PEENING; STRESSES; TENSILE PROPERTIES; ZIRCONIUM ALLOYS

Broader Terms: ALLOYS; COLD WORKING; FABRICATION; HEAT TREATMENTS; JOINING; MATERIALS WORKING; MECHANICAL PROPERTIES; SURFACE TREATMENTS; ZONES

35/7,DE/7 (Item 2 from file: 103)

DIALOG(R)File 103:Energy SciTec

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03891248 JPN-95-008503; EDB-95-135016

Title: Production process for nuclear fuel cladding tube

Author(s)/Editor(s): Hisatsune, Yoshimi; Higashinakagawa, Emiko; Arai, Shinji; Ikeda, Tadahiro.

Corporate Source: Toshiba Corp., Kawasaki, Kanagawa (Japan)

Patent No.: JP 7-109554 A

Patent Assignee(s): Toshiba Corp., Kawasaki, Kanagawa (Japan)

Priority No.: JP 5-276000

Patent Date Filed: 8 Oct 1993

Publication Date: 25 Apr 1995

(6 p)

Language: Japanese

Availability: Available from JAPIO. Also available from EPO.

Abstract: A Zr liner layer is formed on the inner surface of an external tube comprising a Zr alloy, then the external tube and the Zr liner layer are rapidly heated to a high temperature of [alpha] region kept for a short period of time, and then immediately quenched. With such procedures, there can be attained a long-life nuclear fuel cladding tube with excellent uniform corrosion resistance also in the liner portion on the inner side of the Zr alloy tube, and less degradation even upon long time use in a reactor atmosphere. (T.M.).

Major Descriptors: \*CORROSION RESISTANCE -- QUENCHING; \*CORROSION RESISTANCE -- TEMPERATURE DEPENDENCE; \*NUCLEAR FUELS -- FUEL CANS; \*ZIRCONIUM ALLOYS -- NODULAR CORROSION

Descriptors: ENVIRONMENT; LIFETIME; PIPES; WATER VAPOR

Broader Terms: ALLOYS; CHEMICAL REACTIONS; CORROSION; ENERGY SOURCES; FLUIDS; FUELS; GASES; MATERIALS; REACTOR MATERIALS; VAPORS

35/7,DE/8 (Item 3 from file: 103)

DIALOG(R)File 103:Energy SciTec

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03434581 JPN-92-011448; EDB-93-013457

Title: Cladding tube for nuclear fuel and manufacturing method thereof

Author(s)/Editor(s): Oe, Akira.

Corporate Source: Nuclear Fuel Industries Ltd., Tokyo (Japan)

Patent No.: JP 4-128687 A

Patent Assignee(s): Nuclear Fuel Industries Ltd., Tokyo (Japan)



Priority No.: JP 2-248908

Patent Date Filed: 20 Sep 1990

Publication Date: 30 Apr 1992

(7 p)

Language: Japanese

Availability: Available from JAPIO. Also available from INPADOC.

Abstract: A fuel rod cladding tube of a PWR type reactor comprises Sn: 0.9 to 1.2 wt%, Fe: 0.24 to 0.30 wt%, Cr: 0.13 to 0.19 wt%, Nb: 0.05 to 0.15 wt%, Ni: 0.005 to 0.020 wt%, O: 1000 to 1500 ppm, C: 100 to 200 ppm, Si: 50 to 200 ppm and the balance of a zirconium alloy made of Zr and inevitable impurities. Upon fabricating the alloy into a tube, an  $\sigma$  value at the inner surface of the cladding tube is controlled to 0.65 to 0.75, while setting the fabrication degree in the final cold working step, for example, to 60 to 70%. Further, an annealing index is controlled to  $2 \times 10^{-18} [\leq] [\Sigma] \text{Ai} [\leq] 5 \times 10^{-17}$  in an annealing step. With such procedures, the corrosion resistance is improved by decreasing the amount of Sn and adding a trace amount of Ni, in addition, corrosion resistance is also improved by adding a trace amount of Nb. At the same time, hydrogen absorption is also suppressed. Further, corrosion resistance is ensured by adding a greater amount of Si and applying annealing even if the annealing index is low. (T.M.).

Major Descriptors: \*FUEL ASSEMBLIES -- FUEL CANS; \*FUEL ASSEMBLIES -- FUEL RODS; \*PWR TYPE REACTORS -- FUEL ASSEMBLIES

Descriptors: ABSORPTION; ANNEALING; COLD WORKING; CORROSION RESISTANCE; HYDROGEN; ROLLING; TIME DEPENDENCE; TIN ADDITIONS; ZIRCONIUM ALLOYS

Broader Terms: ALLOYS; ELEMENTS; ENRICHED URANIUM REACTORS; FABRICATION; FUEL ELEMENTS; HEAT TREATMENTS; MATERIALS WORKING; NONMETALS; POWER REACTORS; REACTOR COMPONENTS; REACTORS; SORPTION; THERMAL REACTORS; TIN ALLOYS; WATER COOLED REACTORS; WATER MODERATED REACTORS

35/7,DE/9 (Item 4 from file: 103)

DIALOG(R)File 103:Energy SciTec

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03285168 JPN-92-002426; EDB-92-047925

Title: Method for manufacturing composite cladding tube for nuclear fuel

Author(s)/Editor(s): Abe, Hideaki; Kobayashi, Toshimi.

Corporate Source: Sumitomo Metal Industries Ltd., Osaka (Japan)

Patent No.: JP 3-199995 A

Patent Assignee(s): Sumitomo Metal Industries Ltd., Osaka (Japan)

Priority No.: JP 1-341685

Patent Date Filed: 27 Dec 1989

Publication Date: 30 Aug 1991

(6 p)

Language: In Japanese

Availability: Available from JAPIO. Also available from INPADOC.

Abstract: In the present invention, a composite cladding tube for nuclear fuels having a double structure comprising a pure zirconium metal layer joined to the inner surface of a zirconium alloy tube is manufactured at high quality. That is, an intermediate annealing conducted during the repeating cold rolling fabrication for a zirconium alloy tube having a pure zirconium metal layer joined at the inner surface is conducted at a temperature of 450 to 580degC, which is kept for 1 to 4 hours, while the last annealing conducted after completion of the final cold rolling fabrication is conducted at temperature of 550 to 585degC, which is kept for 1 to 4 hours. Alternatively, annealing is applied to a zirconium alloy tube having a pure zirconium metal layer joined at the inner surface at 450 to 580degC for 1 to 4 hours and, thereafter,

it is subjected to cold rolling fabrication. According to the manufacturing method of the present invention, fine defects are not caused to the pure zirconium metal layer at the inner surface, or cause, if any, only shallow defects that can be eliminated.

Accordingly, the production yield can be improved. (I.S.).

Major Descriptors: \*FUEL CANS -- FABRICATION

Descriptors: ANNEALING; BWR TYPE REACTORS; COLD WORKING; MECHANICAL

PROPERTIES; NUCLEAR FUELS; ROLLING; ZIRCONIUM; ZIRCONIUM BASE ALLOYS

Broader Terms: ALLOYS; ELEMENTS; ENERGY SOURCES; ENRICHED URANIUM REACTORS;

FABRICATION; FUELS; HEAT TREATMENTS; MATERIALS; MATERIALS WORKING;

METALS; POWER REACTORS; REACTOR MATERIALS; REACTORS; THERMAL REACTORS;

TRANSITION ELEMENTS; WATER COOLED REACTORS; WATER MODERATED REACTORS;

ZIRCONIUM ALLOYS

35/7,DE/10 (Item 5 from file: 103)

DIALOG(R)File 103:Energy SciTec

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02011080 EDB-87-139033

Author(s): Inagaki, M.; Akabori, K.; Nakajima, J.

Title: Nuclear fuel cladding tube and its manufacture

Patent No.: JP 61-233391 A

Patent Assignee(s): Hitachi Ltd., Tokyo, Japan

Patent Date Filed: Filed date 9 Apr 1985

Publication Date: 17 Oct 1986

p 4

Note: JP patent application 60-74904

Language: Japanese

Abstract: Purpose: To overcome the problems of damage, particularly, stress corrosion crackings due to interactions between fuel cladding tubes and nuclear fuels thereby making the tubes durable for a long period of use. Method: A liner layer comprising a zirconium-based alloy is metallurgically joined to the surface of a tube made of a zirconium-based alloy. The liner layer is applied with a heat treatment of heating to a temperature range for (..cap alpha.. + ..beta..) phase, forming the ..beta.. phase at least to a portion in the ..cap alpha.. phase crystal grain boundary followed by quenching. In this way, the crystal grain boundary of the liner layer is formed as a super-saturated solid-solution phase to obtain a layer in which no intermetallic compound is present or the amount thereof is reduced in the crystal grain boundary. Such a fuel cladding tube is highly reliable against destruction, shows no risk for the tube damage even at an increased burnup degree, by which improvement can be made for increasing the reactor operation cycle and utilizing efficiency.

Major Descriptors: \*FUEL CANS -- DESIGN; \*FUEL CANS -- FABRICATION

Descriptors: ANNEALING; CHROMIUM ADDITIONS; FUEL-CLADDING INTERACTIONS;

GRAIN BOUNDARIES; INTERMETALLIC COMPOUNDS; LINERS; ZIRCONIUM BASE ALLOYS

Broader Terms: ALLOYS; CHROMIUM ALLOYS; CRYSTAL STRUCTURE; HEAT TREATMENTS; MICROSTRUCTURE; ZIRCONIUM ALLOYS

35/7,DE/11 (Item 6 from file: 103)

DIALOG(R)File 103:Energy SciTec

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01819494 AIX-17-059727; EDB-86-143368

Author(s): Takase, Iwao; Yoshida, Toshimi; Ikeda, Shinzo; Masaoka, Isao; Nakajima, Junjiro

Title: Nuclear fuel cladding tube and its fabrication

Corporate Source: Hitachi Ltd., Tokyo (Japan)

Patent No.: JP 60-165580 A

Patent Assignee(s): Hitachi Ltd., Tokyo, Japan

Patent Date Filed: Filed date 8 Feb 1984

Publication Date: 28 Aug 1985

p 10

Note: JP patent application 59-19980

Language: Japanese

Availability: JAPIO. Also available from INPADOC.

Abstract: The purpose of this patent is to manufacture fuel cladding tube made of zirconium-based alloy resistant to nodular corrosions water and steams at high temperature and less sensitive to stress corrosion cracks due to iodine or the like. In light water or heavy water reactor fuels, there are problems such as reduction tube wall thickness caused by the corrosion to the outside and stress corrosion cracks caused by the emission gas and pellet sintering to the inside of fuel cladding tubes. In order to increase the resistance to them, heating upon hardening to ..beta.. phase or (..beta.. + ..cap alpha..) phase region is carried out while cooling the inner surface of the tube by rapid quenching so as to avoid the hardening to the inner surface of the tube. After the hot plastic fabrication, annealing is carried out while making the temperature slope between the inside and the outside of the pipe such that the inside of the fuel cladding tube is at a temperature higher than the re-crystallization point of the alloy and the inside of the can is at a temperature lower than the re-crystallization point. In this method, the outer surface of the fuel cladding tube has a texture having hardened structure and the inner surface has a completely re-crystallized texture, thereby providing excellent nodular corrosion resistance and stress corrosion crack resistance.

Major Descriptors: \*FUEL RODS -- CORROSION RESISTANCE; \*FUEL RODS -- FABRICATION

Descriptors: ANNEALING; CRACKS; PITTING CORROSION; QUENCH HARDENING; RECRYSTALLIZATION; STEAM; STRESS CORROSION; WATER; ZIRCONIUM BASE ALLOYS

Broader Terms: ALLOYS; CHEMICAL REACTIONS; CORROSION; FUEL ELEMENTS; HARDENING; HEAT TREATMENTS; HYDROGEN COMPOUNDS; OXYGEN COMPOUNDS; REACTOR COMPONENTS; ZIRCONIUM ALLOYS

35/7,DE/12 (Item 7 from file: 103)

DIALOG(R)File 103:Energy SciTec

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01409701 AIX-15-028538; EDB-84-107501

Title: New production facilities for nuclear fuel cladding tubes

Author(s): Konishi, Takao; Inoue, Mamoru; Matsuda, Katsuhiko; Kojima, Tatsuhisa (Sumitomo Metal Industries Ltd., Amagasaki, Hyogo (Japan). Steel Tube Works)

Source: Sumitomo Kinzoku (Japan) v 34:1. Coden: SUKIA

Publication Date: Jan 1982

p 161-167

Language: Japanese

Abstract: In order that we may meet the growing demand for nuclear fuel cladding tubes for nuclear power generation, we established the new Zircaloy and stainless steel fuel cladding tube plant on the premises of our Steel Tube Works. The new plant named 'Precision Tube Making Plant' is equipped with a complete series of highly advanced facilities for tube making, finishing and inspecting, and its operation environment is especially kept clean to make severe quality control. In

September 1980, the new plant started operating successfully. Although its present production capacity is 300,000 - 400,000 m/year, we can expand the capacity to the scale of 1,000,000 m/year. This paper gives outlines of manufacturing process and of main manufacturing and inspection equipment for nuclear fuel cladding tubes.;

Major Descriptors: \*FUEL CANS -- INDUSTRIAL PLANTS; \*FUEL CANS -- MANUFACTURING

Descriptors: CLADDING; COLD WORKING; EQUIPMENT; FLOWSHEETS; FUEL FABRICATION PLANTS; INSPECTION; JAPAN; QUALITY CONTROL; STAINLESS STEELS; TUBES; ZIRCALLOY

Broader Terms: ALLOYS; ASIA; CHROMIUM ALLOYS; CONTROL; CORROSION RESISTANT ALLOYS; DEPOSITION; DIAGRAMS; FABRICATION; IRON ALLOYS; IRON BASE ALLOYS; MATERIALS WORKING; NUCLEAR FACILITIES; STEELS; SURFACE COATING; TIN ALLOYS; ZIRCONIUM ALLOYS; ZIRCONIUM BASE ALLOYS

35/7,DE/13 (Item 8 from file: 103)

DIALOG(R)File 103:Energy SciTec

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00638258 AIX-11-514265; EDB-80-077783

Author(s): Shirokane, M.; Chida, T.; Shirai, H.; Watanabe, K.

Title: Method of fabricating fuel rod end plug (Patent)

Patent No.: JP 54-47085 A

Patent Assignee(s): Toshiba Corp., Kawasaki, Kanagawa (Japan)

Publication Date: 13 Apr 1979

p 3

Language: Japanese

Abstract: Purpose: To increase the reliability on the weldability of nuclear fuel pellets with the cladding tube thereby to improve the end plug characteristic. Method: A columnar material made of a zirconium alloy is inserted into a die, and applied with pressure by a punch from the upper part of the columnar material in the heating state, the end part of said columnar material being subjected to extraction forging in the ..cap alpha.. region. By this operation, an end plug whose small diameter portion to be fitted in the fuel rod supporting tool has been shaped is fabricated. Thus, there is produced no residual strain due to the processing at the head of the end plug which is the largest caliber portion, and hence the desired purpose can be achieved.;

Major Descriptors: \*FUEL PELLETS -- WELDABILITY; \*ZIRCALLOY 2 -- WELDABILITY  
Descriptors: CLOSURES; FABRICATION; FORGING; FUEL CANS; FUEL ELEMENTS

Broader Terms: ALLOYS; CHROMIUM ADDITIONS; CHROMIUM ALLOYS; FABRICATION; IRON ADDITIONS; IRON ALLOYS; MATERIALS WORKING; PELLETS; REACTOR COMPONENTS; TIN ALLOYS; ZIRCALLOY; ZIRCONIUM ALLOYS; ZIRCONIUM BASE ALLOYS

35/7,DE/14 (Item 1 from file: 347)

DIALOG(R)File 347:JAPIO

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06082244

MANUFACTURE OF PIPE FOR CLADDING NUCLEAR FUEL

PUB. NO.: 11-023758 [JP 11023758 A]

PUBLISHED: January 29, 1999 (19990129)

INVENTOR(s): ABE HIDEAKI  
TAKEDA KIYOKO

APPLICANT(s): SUMITOMO METAL IND LTD

APPL. NO.: 09-182291 [JP 97182291]

FILED: July 08, 1997 (19970708)

ABSTRACT

PROBLEM TO BE SOLVED: To reduce manufacturing costs by performing intermediate cold rolling working once to a material pipe having a recrystallization structure under specific conditions and performing softening heat treatment, final cold rolling, and annealing heat treatment to the formed object under specific conditions.

SOLUTION: A pipe for cladding nuclear fuel consisting of zirconium based alloy being used for a boiling water type nuclear reactor is manufactured of a solid material pipe and a double material pipe in hot extrusion. For example, the material pipes that have, for example, an outer diameter of approximately 63.5 mm and a thickness of approximately 10.9 mm are subjected to recrystallization annealing heat treatment under specific conditions and are subjected to intermediate cold rolling machining with a section reduction rate to be 90% or higher. Then, softening annealing at 540-680°C, final cold rolling machining with a section reduction rate of 60-85%, and final annealing heat treatment at 550-600°C are performed, thus obtaining a pipe for cladding nuclear fuel with improve mechanical property, where the outer diameter is approximately 11-13 mm and the thickness is approximately 0.7-0.9 mm. The intermediate cold rolling machine and softening annealing with large treatment costs are performed only once, thus reducing manufacturing costs.

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35/7,DE/15 (Item 2 from file: 347)  
DIALOG(R)File 347:JAPIO  
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05990646

ZIRCONIUM ALLOY EXCELLENT IN COLD WORKABILITY AND CORROSION RESISTANCE,  
DUPLEX TUBE FOR CLADDING NUCLEAR FUEL USING THIS ALLOY AND PRODUCTION  
THEREOF

PUB. NO.: 10-273746 [JP 10273746 A]  
PUBLISHED: October 13, 1998 (19981013)  
INVENTOR(s): ABE HIDEAKI  
APPLICANT(s): SUMITOMO METAL IND LTD [000211] (A Japanese Company or  
Corporation), JP (Japan)  
APPL. NO.: 09-013658 [JP 9713658]  
FILED: January 28, 1997 (19970128)  
JAPIO CLASS: 12.3 (METALS -- Alloys); 12.2 (METALS -- Metallurgy & Heat  
Treating); 23.1 (ATOMIC POWER -- General)

ABSTRACT

PROBLEM TO BE SOLVED: To obtain a zirconium alloy excellent in cold-workability and corrosion resistance and suitable as the material for nuclear fuel cladding duplex tube of a water-cooled type nuclear reactor by specifying its composition composed of Sn, Fe, Cr and Zr and further incorporated with Ni and Nb according to necessary.

SOLUTION: This is a low Sn-Zr alloy having a composition containing, by weight, 0.30 to 0.70% Sn, 0.20 to 0.25% Fe and 0.10 to 0.15% Cr, furthermore containing, at need, one or both of 0.005 to 0.05% Ni and 0.05 to 0.20% Nb, and the balance Zr with inevitable impurities, and in which cracks and strains are not generated even by cold rolling in which the

reduction of cross-sectional area is regulated to about  $\geq 80\%$ . By forming an outer tube by this Zr alloy and making an inner tube of a high Sn-Zr alloy containing 1.2 to 1.7% Sn, the duplex tube for cladding nuclear fuel excellent in corrosion resistance in the outer face in which CSR value defined by the formula of  $CSR = \frac{\epsilon_s}{\epsilon_r}$  ( $\epsilon_s$  and  $\epsilon_r$  denote the strains in the circumferential direction and to strains in the thickness direction) is equal to that of a solid tube of the Zr alloy same as that of the inner tube and having high strength in the whole body can securely be obtained at a low cost.

35/7,DE/16 (Item 3 from file: 347)  
DIALOG(R)File 347:JAPIO  
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05790590

HIGHLY ANTICORROSIVE CLADDING TUBE FOR NUCLEAR FUEL, SPACER, CHANNEL BOX, FUEL ASSEMBLY THEREOF AND METHOD FOR MANUFACTURING IT

PUB. NO.: 10-073690 [JP 10073690 A]  
PUBLISHED: March 17, 1998 (19980317)  
INVENTOR(s): INAGAKI MASATOSHI  
TAKASE IWAO  
SUGANO MASAYOSHI  
KUNIYA JIRO  
AKAHORI KIMIHIKO  
MASAOKA ISAO  
MAKI HIDEO  
NAKAJIMA JUNJIRO

APPLICANT(s): HITACHI LTD [000510] (A Japanese Company or Corporation), JP (Japan)

APPL. NO.: 09-175505 [JP 97175505]

FILED: July 01, 1997 (19970701)

JAPIO CLASS: 23.1 (ATOMIC POWER -- General); 12.2 (METALS -- Metallurgy & Heat Treating); 12.3 (METALS -- Alloys)

JAPIO KEYWORD: R002 (LASERS); R003 (ELECTRON BEAM)

#### ABSTRACT

PROBLEM TO BE SOLVED: To improve crossion resistance and hydrogen absorption characteristics by composing at least one of a cladding tube for nuclear fuel, a spacer for a nuclear fuel assembly and a channel box for a nuclear fuel assembly of an alloy made of specific materials in specific proportions.

SOLUTION: In a fuel assembly for a nuclear reactor, at least one of a cladding tube, a spacer and a channel box is made of an alloy of the following composition in weight percentage. The alloy contains tin of about 1.20 to 2%, iron of about 0.20 to 0.5%, chromium of about 0.05 to 0.15% and nickel of about 0.03% to 0.16%, with the rest constituted materially of zirconium. These metal elements constitute a zirconium-group alloy where the proportion of iron to nickel is about 1.4 to 10. After hot plastic working, this alloy is heated and maintained for a short time in a .beta. phase or a temperature range containing .alpha. and .beta. phases and then is rapidly cooled before cold working. Subsequently, cool plastic working and anneal working are repeated alternately and shaping work is done by forming a thin-wall material made of a zirconium-group alloy. This makes it possible to obtain an alloy which is excellent in corrosion resistance and absorbs little hydrogen and enables the higher burn-up of fuel.

35/7,DE/17 (Item 4 from file: 347)  
DIALOG(R)File 347:JAPIO  
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04816954

## PRODUCTION OF NUCLEAR FUEL CLADDING PIPE

PUB. NO.: 07-109554 [JP 7109554 A]  
PUBLISHED: April 25, 1995 (19950425)  
INVENTOR(s): HISATSUNE YOSHIMI  
HIGASHINAKAGAHA EMIKO  
ARAI SHINJI  
IKEDA TADAHIRO  
APPLICANT(s): TOSHIBA CORP [000307] (A Japanese Company or Corporation), JP  
(Japan)  
APPL. NO.: 05-276000 [JP 93276000]  
FILED: October 08, 1993 (19931008)  
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC  
POWER -- General)

## ABSTRACT

PURPOSE: To produce a nuclear fuel cladding pipe excellent in uniform corrosion resistance in an external pipe and excellent in corrosion resistance in a steam environment even in a liner layer by providing a special heat treating operation in the producing process.

CONSTITUTION: This method comprises stage in which a Zr liner layer is formed on the inside face of an external pipe constituted of a Zr alloy and a stage in which the external pipe and Zr liner are rapidly heated to a high temperature in an .alpha. region, is held and is thereafter rapidly cooled. Thus, the nuclear fuel cladding pipe in which uniform corrosion resistance is improved without deteriorating its nodular corrosion resistance in the external pipe and corrosion resistance in a steam environment is excellent even in the liner layer can be produced.

35/7,DE/18 (Item 5 from file: 347)  
DIALOG(R)File 347:JAPIO  
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03373348

## CORROSION RESISTING ZIRCONIUM-BASE ALLOY AND ITS PRODUCTION

PUB. NO.: 03-036248 [JP 3036248 A]  
PUBLISHED: February 15, 1991 (19910215)  
INVENTOR(s): EITO YOSHINORI  
APPLICANT(s): NIPPON NUCLEAR FUEL DEV CO LTD [472479] (A Japanese Company  
or Corporation), JP (Japan)  
APPL. NO.: 01-171139 [JP 89171139]  
FILED: July 04, 1989 (19890704)  
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 12.6 (METALS --  
Surface Treatment); 23.1 (ATOMIC POWER -- General)

## ABSTRACT

PURPOSE: To provide nodular corrosion resistance suitable for a core internal structure material by irradiating a film layer consisting of the prescribed elements formed on the surface of the alloy for nuclear reactor with a corpuscular beam and allowing a region deficient in the solid solution components of alloying elements to disappear.

CONSTITUTION: An ingot is prepared by adding the prescribed alloying elements (tin, iron, Cr, Ni, etc.) to pure-Zr sponge for nuclear reactor fuel clad pipe or core internal structure material and carrying out arc melting. The above ingot is subjected to .beta.- forging, solution treatment, and .alpha.-forging and is formed into a tube stock by means of hot working, which is formed into the desired thin-wall finished product while exerting a repetition of cold rolling and annealing. In the above method, after the final annealing stage, a film layer consisting of the prescribed elements (Co, Ni, iron, etc.) is formed by a plating method, to which corpuscular beam irradiation is applied. By this method, the region deficient in the solid solution components of alloying elements in the vicinity of the surface of the Zr-base alloy can be allowed to disappear, and as a result, nodular corrosion resulting from the presence of the above deficient region can be inhibited.

35/7,DE/19 (Item 6 from file: 347)  
DIALOG(R)File 347:JAPIO  
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02039251  
SUPERPLASTIC ZIRCONIUM ALLOY AND ITS MANUFACTURE

PUB. NO.: 61-253351 [JP 61253351 A]  
PUBLISHED: November 11, 1986 (19861111)  
INVENTOR(s): KUBO TOSHIO  
MOTOMIYA TAKEO  
APPLICANT(s): NIPPON NUCLEAR FUEL DEV CO LTD [472479] (A Japanese Company  
or Corporation), JP (Japan)  
APPL. NO.: 60-092821 [JP 8592821]  
FILED: April 30, 1985 (19850430)  
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 12.3 (METALS --  
Alloys); 23.1 (ATOMIC POWER -- General)

#### ABSTRACT

PURPOSE: To manufacture a superplastic Zr alloy having superior resistance to stress corrosion cracking and corrosion by hardening and aging a Zr alloy for the constituent parts of a nuclear fuel assembly for a nuclear fission reactor under specified conditions.

CONSTITUTION: When a pipe 1 for cladding fuel pellets 2 for a nuclear fission reactor is made of a Zr alloy, the pipe 1 causes corrosion by a coolant and stress corrosion cracking by the thermal expansion of the pellets 2. In order to prevent the corrosion and stress corrosion cracking and to provide superplasticity, the Zr alloy is subjected to solution heat treatment in the .alpha.+ .beta. phase region (800-950 deg.C) of the alloy, cooled to 700 deg.C at <=100 deg.C/sec cooling rate, and aged at a proper temperature of 550-700 deg.C in the .alpha.-phase region. By this heat treatment, a cladding pipe of a Zr alloy causing no stress corrosion cracking and having superior resistance to corrosion by a coolant is obtained

35/7,DE/20 (Item 7 from file: 347)  
DIALOG(R)File 347:JAPIO  
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01965760  
MANUFACTURE OF NUCLEAR FUEL CLADDING TUBE MADE OF ZR BASE ALLOY



PUB. NO.: 61-179860 [JP 61179860 A]  
PUBLISHED: August 12, 1986 (19860812)  
INVENTOR(s): ABE HIDEAKI  
HONCHI MASAHIRO  
APPLICANT(s): SUMITOMO METAL IND LTD [000211] (A Japanese Company or Corporation), JP (Japan)  
APPL. NO.: 60-001177 [JP 851177]  
FILED: January 08, 1985 (19850108)  
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC POWER -- General)

## ABSTRACT

PURPOSE: To manufacture stably and surely a pipe superior in anisotropy and mechanical property of material characteristic, by performing finishing cold roll working and annealing under specified conditions, at manufacturing nuclear fuel cladding tube composed of Zr base alloy having a specified component composition

CONSTITUTION: In manufacturing the titled tube composed of, by weight ratio 1.20-1.70% Sn, 0.18-0.24% Fe, 0.07-0.13% Cr, under  $(Fe(\%) + Cr(\%)) = 0.28-0.37$ , and the balance Zr with inevitable impurities, finishing cold roll working is carried out under 1.5-3.5 working parameter (QE value),  $(3.3 + 0.16 \ln QE) / 4.8 \times 100 - 90\%$  working degree Rd. QE value is exhibited by a formula, Dm is average tube diameter before rolling, (dm) is said diameter after rolling, T is tube wall thickness before rolling, (t) is said thickness after rolling. Successive annealing is performed as stress removal annealing at 430-500 deg.C.

35/7,DE/21 (Item 8 from file: 347)  
DIALOG(R)File 347:JAPIO  
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01965759

## MANUFACTURE OF NUCLEAR FUEL CLADDING TUBE MADE OF ZR BASE ALLOY

PUB. NO.: 61-179859 [JP 61179859 A]  
PUBLISHED: August 12, 1986 (19860812)  
INVENTOR(s): ABE HIDEAKI  
HONCHI MASAHIRO  
APPLICANT(s): SUMITOMO METAL IND LTD [000211] (A Japanese Company or Corporation), JP (Japan)  
APPL. NO.: 60-001176 [JP 851176]  
FILED: January 08, 1985 (19850108)  
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC POWER -- General)

## ABSTRACT

PURPOSE: To manufacture stably, surely a tube superior in anisotropy and mechanical property of material characteristic, by performing cold roll working and annealing under specified conditions, at manufacturing nuclear fuel cladding tube composed of Zr base alloy having a specified component composition

CONSTITUTION: In manufacturing the titled tube composed of, by weight ratio 1.20-1.70% Sn, 0.07-0.20% Fe, 0.05-0.15% Cr, 0.03-0.08 Ni under  $(Fe(\%) + Cr(\%) + Ni(\%)) = 0.18-0.38$ , and the balance Zr with inevitable impurities, finishing clf roll working is carried out under 1.5-3.5 working parameter (QE value),  $\leq (6.6 + 0.50 \ln QE) / 8.4 \times 100$  working degree (Rd)(%). QE value is exhibited by a formula, Dm is average tube diameter before rolling, dm is said diameter after rolling, T is tube wall thickness before rolling, (t) is said thickness after rolling. Successive annealing is

performed as recrystallization annealing at  $\geq 550$  deg.C.

35/7,DE/22 (Item 9 from file: 347)  
DIALOG(R)File 347:JAPIO  
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01795464

MANUFACTURE OF COMPOSITE CLAD PIPE FOR NUCLEAR REACTOR

PUB. NO.: 61-009564 [JP 61009564 A]  
PUBLISHED: January 17, 1986 (19860117)  
INVENTOR(s): ASAHI KAZUMI  
APPLICANT(s): NIPPON NUCLEAR FUEL DEV CO LTD [472479] (A Japanese Company  
or Corporation), JP (Japan)  
APPL. NO.: 59-129361 [JP 84129361]  
FILED: June 25, 1984 (19840625)  
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 12.5 (METALS --  
Working); 23.1 (ATOMIC POWER -- General)

ABSTRACT

PURPOSE: To improve the mechanical strength and corrosion resistance by forming a layer of a uniform thickness having a .beta.-hardened structure on the outside of a Zr alloy layer having an .alpha.-tempered structure.

CONSTITUTION: An .alpha.-tempered pierced Zr alloy billet 1 having an .alpha.-tempered structure and a .beta.-hardened pierced billet 2 having a .beta.-hardened structure are formed. The billet 1 is put in the billet 2 and united to one body by hot rolling mill 3 to manufacture a rough pipe 4 for a composite clad pipe. The pipe 4 is cold rolled to prescribed final dimensions to obtain a composite clad pipe for a nuclear reactor consisting of a Zr alloy layer 5 having an .alpha.-tempered structure and a layer 6 of a uniform thickness having a .beta.-hardened structure formed on the outside of the layer 5.

35/7,DE/23 (Item 10 from file: 347)  
DIALOG(R)File 347:JAPIO  
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01743060

MANUFACTURE OF ZIRCONIUM ALLOY

PUB. NO.: 60-221560 [JP 60221560 A]  
PUBLISHED: November 06, 1985 (19851106)  
INVENTOR(s): NAKAJIMA JUNJIRO  
SHINPO KATSUTOSHI  
YASUDA TETSUO  
ASANO RINICHI  
INAGAKI MASATOSHI  
APPLICANT(s): HITACHI LTD [000510] (A Japanese Company or Corporation), JP  
(Japan)  
APPL. NO.: 59-077399 [JP 8477399]  
FILED: April 16, 1984 (19840416)  
JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC  
POWER -- General)

ABSTRACT

PURPOSE: To manufacture a nuclear fuel cladding pipe with high corrosion resistance by quenching a hot worked Zr alloy plate from a specified

temperature and subjecting it to cold plastic working and annealing plural times.

CONSTITUTION: When a fuel cladding pipe for a nuclear power plant is manufactured with a Zr alloy such as zircaloy-2, a Zr alloy ingot is worked into a plate by hot rolling or other process. This plate is hardened by heating to  $\geq 870^{\circ}\text{C}$  in a temperature range in which .alpha.- and .beta.-phase are included, holding at the temperature for  $\geq 30$ sec, and quenching to  $500^{\circ}\text{C}$  at  $\geq 100^{\circ}\text{C/sec}$  cooling rate to form a fine acicular structure consisting of fine grains. The hardened plate is subjected to cold plastic working and annealing plural times to manufacture a nuclear fuel cladding pipe. This Zr alloy cladding pipe prevents nodular corrosion in a nuclear reactor for a long period.

35/7,DE/24 (Item 11 from file: 347)

DIALOG(R)File 347:JAPIO

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01571655

#### PRODUCTION OF NUCLEAR FUEL CLADDING PIPE

PUB. NO.: 60-050155 [JP 60050155 A]

PUBLISHED: March 19, 1985 (19850319)

INVENTOR(s): HIGASHINAKAGAHA EMIKO

SATO KANEMITSU

KAWASHIMA JUNKO

KAMEI TOSHIO

APPLICANT(s): TOSHIBA CORP [000307] (A Japanese Company or Corporation), JP (Japan)

NIPPON ATOM IND GROUP CO LTD [352264] (A Japanese Company or Corporation), JP (Japan)

APPL. NO.: 58-157610 [JP 83157610]

FILED: August 29, 1983 (19830829)

JAPIO CLASS: 12.2 (METALS -- Metallurgy & Heat Treating); 23.1 (ATOMIC POWER -- General)

#### ABSTRACT

PURPOSE: To improve the corrosion resistance and mechanical characteristic of a Zr alloy pipe as a nuclear fuel cladding pipe by subjecting the Zr alloy pipe to a drawing stage until said pipe is reduced to the prescribed bore and wall thickness then heating quickly the pipe to a high temperature in an .alpha. region as a final heat treatment and cooling quickly the heated pipe immediately after holding for a short time.

CONSTITUTION: A hollow billet consisting of a Zr alloy is hot-extruded and is then subjected to a drawing stage consisting of cold working, by which the billet is reduced to the finishing bore and wall thickness. The drawing stage is accomplished in combination with annealing at the intermediate and the final annealing is accomplished with 3-4 passes. The Zr alloy pipe subjected to the final annealing is quickly heated at the surface part up to  $780-860^{\circ}\text{C}$  in the .alpha. region by, for example, high-frequency heating and after the pipe is held for several seconds, the pipe is quickly cooled so that the residual stress remains on the surface part. The nuclear fuel cladding pipe having excellent resistance to nodular corrosion, an excellent mechanical characteristic and less bending and twisting is thus obtained

35/7,DE/25 (Item 12 from file: 347)

DIALOG(R)File 347:JAPIO  
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00395085

## MANUFACTURING METHOD OF END PLUG OF NUCLEAR FUEL ROD

PUB. NO.: 54-047085 [JP 54047085 A]  
PUBLISHED: April 13, 1979 (19790413)  
INVENTOR(s): SHIROKANE MAKOTO  
SENDA CHUICHI  
SHIRAI HIDEO  
WATANABE KUNIMICHI  
APPLICANT(s): TOSHIBA CORP [000307] (A Japanese Company or Corporation), JP  
(Japan)  
APPL. NO.: 52-112531 [JP 77112531]  
FILED: September 21, 1977 (19770921)  
JAPIO CLASS: 23.1 (ATOMIC POWER -- General); 12.5 (METALS -- Working)

## ABSTRACT

PURPOSE: To heighten the reliability of welding with a cladding tube of nuclear fuel pellets, and to improve the characteristic of an end plug, without causing a residual strain due to the processing of the head part of the end plug to be generated, by means of moulding the end part of a circular cylindrical material of Zr-series alloy by an extrusion precision casting in .alpha. zone.

CONSTITUTION: A circular cylindrical material 10 of Zr-series alloy is inserted in a punch 11 and a dies 12, and pressure is exerted in a heated state from the upper part of the circular cylindrical material 10, thereby an end plug 4 is manufactured by means of forming the end part of the circular cylindrical material 10 into the small diameter part 5 which is engaged with a supporting tool of a nuclear fuel by an extrusion precision casting in .alpha. zone. A residual strain due to the processing of the head part 6 which is the max. aperture part of the end plug 4, does not been generated, thereby, the reliability of welding an fixing with the cladding tube into which the fuel pellets is inserted, is heightened, and the characteristic of the end plug can be improved

35/7,DE/26 (Item 1 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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011247439

WPI Acc No: 1997-225342/199720

Zirconium alloy tubing fabrication used for BWR and PWR - by coarsening annealing alloy billet, forming into tube, heat treating outside of tube while cooling inside, then rapidly quenching outside of tube giving crack and corrosion resistance

Patent Assignee: GENERAL ELECTRIC CO (GENE )

Inventor: ADAMSON R B; POTTS G A

Number of Countries: 001 Number of Patents: 001

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
US 5618356	A	19970408	US 9352791	A	19930423	199720 B
			US 9352793	A	19930423	
			US 95489597	A	19950612	

Priority Applications (No Type Date): US 95489597 A 19950612; US 9352791 A 19930423; US 9352793 A 19930423

## Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
US 5618356	A	13		C22F-001/18	CIP of application US 9352791 CIP of application US 9352793 CIP of patent US 5437747 CIP of patent US 5519748

## Abstract (Basic): US 5618356 A

Preparing a corrosion resistant Zr alloy tube comprises coarsening annealing at 621 deg. C, converting alloy billet to alloy tube, heat treating an outer region of the tube at alpha + beta temp. region while cooling inner region of the tube (104), rapidly quenching outer region and performing one or more cold working steps, each cold work step being followed by an anneal step at a temp of 576 deg. C.

Further claimed is a process similar to the above where the cold work steps are dispersed throughout the process. Also claimed is a process for forming a barrier containing tube similar to the above.

USE - The zircalloy cladding is used in the nuclear fuel industry, particularly for BWR and PWR

ADVANTAGE - Tube is resistant to axial crack propagation, crack initiation and nodular corrosion.

Dwg.3/5

Title Terms: ZIRCONIUM; ALLOY; TUBE; FABRICATE; BWR; PWR; COARSE; ANNEAL; ALLOY; BILLET; FORMING; TUBE; HEAT; TREAT; TUBE; COOLING; RAPID; QUENCH; TUBE; CRACK; CORROSION; RESISTANCE

Derwent Class: K05; M29; X14

International Patent Class (Main): C22F-001/18

35/7,DE/27 (Item 2 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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010750189

WPI Acc No: 1996-247144/199625

Corrosion-resistant zirconium alloy mfr. - by beta-treatment, hot working, repeatedly cold working and process annealing, e.g. used for LWR nuclear fuel cladding tubes

Patent Assignee: SUMITOMO METAL IND LTD (SUMQ )

Number of Countries: 001 Number of Patents: 001

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
JP 8100231	A	19960416	JP 94236512	A	19940930	199625 B

Priority Applications (No Type Date): JP 94236512 A 19940930

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
JP 8100231	A	5		C22C-016/00	

## Abstract (Basic): JP 8100231 A

The Zr-alloy is made by hot working after beta-treatment, repeatedly cold working and process annealing, and final annealing after final cold working. The process annealing before final cold working is performed at temps. of at least 750deg.C.

USE - Used for fuel cladding tubes of light water reactors.

ADVANTAGE - Good resistance to general corrosion.

Dwg.0/1

Title Terms: CORROSION; RESISTANCE; ZIRCONIUM; ALLOY; MANUFACTURE; BETA; TREAT; HOT; WORK; REPEAT; COLD; WORK; PROCESS; ANNEAL; NUCLEAR; FUEL; CLAD; TUBE

Index Terms/Additional Words: LIGHT; WATER; REACTOR  
Derwent Class: K05; M26; X14  
International Patent Class (Main): C22C-016/00  
International Patent Class (Additional): C22F-001/18

35/7,DE/28 (Item 3 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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010699819

WPI Acc No: 1996-196774/199620

High corrosion resistant zirconium alloy prodn. - by soln. treating  
zirconium alloy, hot and/or cold working, and heat treatment at specific  
range, used to clad tubes of nuclear power plants

Patent Assignee: SUMITOMO METAL IND LTD (SUMQ )

Number of Countries: 001 Number of Patents: 001

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
JP 8067954	A	19960312	JP 94200871	A	19940825	199620 B

Priority Applications (No Type Date): JP 94200871 A 19940825

Patent Details:

Patent No	Kind	Lan Pg	Main IPC	Filing Notes
JP 8067954	A	6	C22F-001/18	

Abstract (Basic): JP 8067954 A

The Zr-alloy is made by soln. treating a Zr-alloy stock contg. (by wt.) 0.4-1.7% Sn, 0.25-0.75% Fe, 0.05-0.30% Cr, 0-0.10% Ni, and 0-1.0% Ni, followed by hot working and/or cold working, in which heat treatment at alpha-phase range accompanied with the working is performed in a range where the total value of heat treatment parameter ( $A_i$ ) in each heat treatment is  $8.5 \times 10^{-16}$ - $2.1 \times 10^{-14}$ , (where  $A_i = t_i \times \exp(-650/RT_i)$ ,  $t_i$  = heat treating time in i-th heat treating process, and  $T_i$  = heat treating temps. in i-th heat treating process, and  $R$  = gas constant (cal/mol.K)).

USE - For fuel cladding tubes of nuclear power plants.

Dwg.0/0

Title Terms: HIGH; CORROSION; RESISTANCE; ZIRCONIUM; ALLOY; PRODUCE;  
SOLUTION; TREAT; ZIRCONIUM; ALLOY; HOT; COLD; WORK; HEAT; TREAT; SPECIFIC  
; RANGE; CLAD; TUBE; NUCLEAR; POWER; PLANT

Derwent Class: K05; M26; M29

International Patent Class (Main): C22F-001/18

International Patent Class (Additional): C22C-016/00

35/7,DE/29 (Item 4 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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010436608

WPI Acc No: 1995-337924/199544

Nuclear fuel cladding tube mfr. - involves annealing to diffuse alloying  
elements from lining into zirconium@ barrier layer

Patent Assignee: GENERAL ELECTRIC CO (GENE )

Inventor: ADAMSON R B; ARMIJO J S; LUTZ D R

Number of Countries: 004 Number of Patents: 006

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
DE 19509049	A1	19950928	DE 1009049	A	19950314	199544 B

SE 9500969	A	19950922	SE 95969	A	19950320	199549
US 5469481	A	19951121	US 9391672	A	19930714	199601
			US 94215456	A	19940321	
JP 8043567	A	19960216	JP 9560179	A	19950320	199617
JP 2815551	B2	19981027	JP 9560179	A	19950320	199848
SE 510099	C2	19990419	SE 95969	A	19950320	199922

Priority Applications (No Type Date): US 94215456 A 19940321; US 9391672 A 19930714

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
DE 19509049	A1		9	G21C-021/02	
US 5469481	A		9	G21C-003/00	CIP of application US 9391672 CIP of patent US 5383228
JP 8043567	A		8	G21C-003/06	
JP 2815551	B2		8	G21C-003/06	Previous Publ. patent JP 8043567
SE 9500969	A			G21C-003/20	
SE 510099	C2			G21C-003/20	

Abstract (Basic): DE 19509049 A

Prod'n. of a cladding tube, with an outer substrate, an intermediate Zr barrier layer and an inner alloyed Zr lining, involves (a) joining the outer surface of the barrier layer to the inner surface of the substrate; (b) joining the outer surface of the inner lining to the inner surface of the barrier layer; and (c) diffusion annealing to diffuse alloying elements from the lining into the barrier layer for forming a diffusion layer having an alloying element concn. which decreases from the inner surface of the barrier layer to a zero concn. location within the barrier layer. Also claimed is a similar process in which steps (a)-(c) are followed by (d) cold working with intermediate stress-relief or recrystallisation anneals; and (e) heating no more than the outer 33% of the outer substrate in the alpha & beta-phases or the beta-phase region and cooling to produce a distribution of fine pptes. in the outer region of the substrate.

USE - As a nuclear reactor fuel cladding tube.

ADVANTAGE - The barrier layer is alloyed at its inner surface (facing the fuel), to provide corrosion resistance, and is unalloyed at its outer region to retain sufficient ductility to protect against damage caused by fuel pellet-cladding interaction.

Dwg.2/2

Abstract (Equivalent): US 5469481 A

A method of making a cladding tube having an outer substrate, an intermediate zirconium barrier layer, and a zirconium-based inner liner having alloying elements, the substrate, barrier layer, and inner liner each having interior and exterior circumferential surfaces, the method comprising the following steps:

(a) bonding the zirconium barrier layer exterior circumferential surface to the substrate interior circumferential surface;

(b) bonding the inner liner outer circumferential surface to the zirconium barrier layer inner circumferential surface; and

(c) conducting a diffusion anneal after steps (a) and (b) at a time and temperature sufficient to cause the alloying elements from the inner liner to diffuse into the barrier layer to form a diffusion layer containing a concentration of alloying elements that decreases from the interior circumferential surface of the barrier layer to a location interior to the barrier layer where there is substantially no alloying elements, wherein the alloying elements in the diffusion layer impart corrosion resistance to the barrier layer.

Dwg.0/2

Title Terms: NUCLEAR; FUEL; CLAD; TUBE; MANUFACTURE; ANNEAL; DIFFUSION;

ALLOY; ELEMENT; LINING; ZIRCONIUM; BARRIER; LAYER

Derwent Class: K05; M26; P71; X14

International Patent Class (Main): G21C-003/00; G21C-003/06; G21C-003/20;  
G21C-021/02

International Patent Class (Additional): B30B-012/00; G21D-001/00

35/7,DE/30 (Item 5 from file: 351)

DIALOG(R)File 351:Derwent WPI

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010066940

WPI Acc No: 1994-334653/199442

Fabrication of zirconium alloy cladding tube - produces a component which has a high resistance to crack propagation.

Patent Assignee: GENERAL ELECTRIC CO (GENE )

Inventor: ADAMSON R B; POTTS G A

Number of Countries: 010 Number of Patents: 006

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
EP 622470	A1	19941102	EP 94302539	A	19940411	199442 B
TW 233361	A	19941101	TW 94100032	A	19940104	199503
JP 7090522	A	19950404	JP 9481201	A	19940420	199522
US 5437747	A	19950801	US 9352791	A	19930423	199536
US 5681404	A	19971028	US 9352791	A	19930423	199749
			US 95385807	A	19950209	
MX 186392	B	19971013	MX 942968	A	19940422	199901

Priority Applications (No Type Date): US 9352791 A 19930423; US 95385807 A 19950209

Cited Patents: EP 425465; US 4576654; US 4671826; US 4718949

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
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EP 622470	A1	E	14	C22F-001/18	
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Designated States (Regional): CH DE ES IT LI SE

TW 233361	A			G21C-003/00	
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JP 7090522	A		11	C22F-001/18	
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US 5437747	A		13	C22F-001/18	
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US 5681404	A		11	C22F-001/18	
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Div ex application US 9352791

Div ex patent US 5437747

MX 186392	B			C22F-001/018	
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Abstract (Basic): EP 622470 A

A method of processing a zirconium alloy tube is disclosed. A coarsening anneal is performed at 621 deg. C. min. for 1-100 hours and an outer region of the tube is selectively heat treated to at least the alpha plus beta region whilst cooling the inner region of the tube and rapidly quenching the outer region. One or more cold work operations are carried out each followed by an annealing heat treatment at 576 deg. C. Min.

The zirconium alloy is pref. selected from Zircalloy-2, Zircalloy-4 and Zirloy. The selective heat-treatment step heats the outer region of the tube to 927 deg. C. approx. The accumulated normalised annealing time is approx. 5 multiply 10-17 hours. The outer region may be selectively heated by an induction coil heater.

USE -For fabricating zirconium alloy cladding for use in nuclear fuel elements.

ADVANTAGE - The cladding is resistant to axial crack propagation, crack initiation and corrosion.

Dwg.3/4



Abstract (Equivalent): US 5681404 A

A cladding tube having precipitates that vary in coarseness and density across the cladding wall, with coarse precipitates having an average diameter ranging from between about 0.15 and 2 micrometers in an inner region and fine precipitates having an average diameter ranging from between about 0.01 and 0.15 micrometers in an outer region, the cladding tube fabricated from a zirconium alloy tube by a method comprising steps of:

(a) performing a coarsening anneal on the zirconium alloy tube at a temperature of at least about 700 deg. C. for between about 1 and 100 hours such that precipitates coarsen throughout the entire tube; (b) selectively heat treating the outer region of the zirconium alloy tube by first heating the outer region to at least the alpha plus beta region while cooling the inner region of the tube and then rapidly quenching the outer region; and (c) performing one or more cold work steps on the zirconium alloy tube, each followed by an annealing step, the annealing step or steps being conducted at a temperature of greater than about 576 deg. C. , where the coarse precipitates in the inner region impart resistance to axial crack propagation in the cladding tube.

Dwg.3/3

US 5437747 A

Zirconium alloy cladding tube is fabricated by a method in which the tube is subjected to an anneal at a temp. above 621 deg.C, pref. at 775 deg.C for 4 hr. to coarsen pptes. throughout the tube. An outer region is selectively heated, pref. in an induction coil, to at least the alpha plus beta region, while the interior is cooled, and the outer region is then quenched. One or more cold work steps are each followed by an anneal at a temp. greater than 576 deg.C. pref. at 620-650 deg.C.

USE/ADVANTAGE - Used in nuclear fuel elements. Tube is resistant to axial crack propagation.

(Dwg.0/4)

Title Terms: FABRICATE; ZIRCONIUM; ALLOY; CLAD; TUBE; PRODUCE; COMPONENT; HIGH; RESISTANCE; CRACK; PROPAGATE

Derwent Class: K05; M26; M29; X14

International Patent Class (Main): C22F-001/018; C22F-001/18; G21C-003/00

International Patent Class (Additional): G21C-003/06; G21C-003/07

35/7,DE/31 (Item 6 from file: 351)

DIALOG(R)File 351:Derwent WPI

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009967240

WPI Acc No: 1994-234953/199428

Mfr. of nuclear fuel elements having fuel rods and cladding tubes - where the zirconium cladding tube has a zirconium@ alloy internal liner whose temp. during mfr. does not exceed the temp. when an incipient phase transformation to beta phase takes place.

Patent Assignee: ABB ATOM AB (ALLM )

Inventor: DAHLBAECK M; DAHLBACK M

Number of Countries: 020 Number of Patents: 010

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
WO 9415343	A1	19940707	WO 93SE1070	A	19931215	199428 B
FI 9502981	A	19950616	WO 93SE1070	A	19931215	199538
			FI 952981	A	19950616	
EP 674800	A1	19951004	WO 93SE1070	A	19931215	199544
			EP 94903202	A	19931215	

JP 8505225	W	19960604	WO 93SE1070	A	19931215	199648
			JP 94515076	A	19931215	
EP 674800	B1	19970326	WO 93SE1070	A	19931215	199717
			EP 94903202	A	19931215	
US 5620536	A	19970415	WO 93SE1070	A	19931215	199721
			US 94284648	A	19940811	
DE 69309305	E	19970430	DE 609305	A	19931215	199723
			WO 93SE1070	A	19931215	
			EP 94903202	A	19931215	
ES 2102810	T3	19970801	EP 94903202	A	19931215	199737
SE 506174	C2	19971117	SE 923871	A	19921218	199801
JP 3031714	B2	20000410	WO 93SE1070	A	19931215	200023
			JP 94515076	A	19931215	

Priority Applications (No Type Date): SE 923871 A 19921218

Cited Patents: EP 121204; EP 155603; EP 194797; EP 425465; SE 459340

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
WO 9415343	A1	E	21	G21C-021/02	
Designated States (National): FI JP US					
Designated States (Regional): AT BE CH DE DK ES FR GB GR IE IT LU MC NL PT SE					
JP 3031714	B2		7	G21C-003/20	Previous Publ. patent JP 8505225
					Based on patent WO 9415343
EP 674800	A1	E		G21C-021/02	Based on patent WO 9415343
Designated States (Regional): CH DE ES LI SE					
JP 8505225	W		21	G21C-003/20	Based on patent WO 9415343
EP 674800	B1	E	8	G21C-021/02	Based on patent WO 9415343
Designated States (Regional): CH DE ES LI SE					
US 5620536	A		5	G21C-021/02	Based on patent WO 9415343
DE 69309305	E			G21C-021/02	Based on patent EP 674800
					Based on patent WO 9415343
ES 2102810	T3			G21C-021/02	Based on patent EP 674800
FI 9502981	A			G21C-000/00	
SE 506174	C2			G21C-003/20	

Abstract (Basic): WO 9415343 A

Method of mfr. of a nuclear fuel element comprising a composite cladding tube with a Zr or Zr alloy inner component suitable for PCI resistant composite cladding. The composite cladding has an outer Zr alloy component which constitutes the supporting part of the composite cladding tube, e.g. Zircalloy 2, Zircalloy 4, or Zr-2.5Nb. The cladding tube is mfd. by fabricating an ingot of the compsn. of the inner component and an ingot of the compsn. of the outer component, respectively. The ingots are separately machined into a billet of a suitable dimension. The ingots are joined together and extruded to a tube blank. The blank is further machined by cold rolling and intermediate heat treatment operations, and a final heat treatment in the final dimensions.

The mfr. of the inner component from the ingot prodn. up to completion of the cladding tube, comprises forging, rolling, extrusion, heat treatment and final heat treatment.

USE/ADVANTAGE - Method of mfr. of nuclear fuel elements comprising fuel rods whose cladding tubes are provided with Zr or Zr alloy internal liner. Nuclear fuel elements has improved resistance to the corrosive effect of water and steam in case of damage.

Dwg.1/1

Abstract (Equivalent): EP 674800 B

A method for manufacturing a nuclear fuel element comprising a composite cladding tube with an inner component of zirconium or a

zirconium alloy suitable as inner component in a PCI-resistant composite cladding as well as an outer component of a zirconium alloy intended to constitute the supporting part of a composite cladding tube, such as, for example, Zircaloy 2, zircaloy 4 or Zr 2.5 Nb, wherein the cladding tube is manufactured by fabricating an ingot of the composition of the inner component and an ingot of the composition of the outer component, respectively, and machining them separately into a billet of a suitable dimension and thereafter joining them together and extruding to a tube blank and machining it further by means of cold rolling and intermediate heat-treatment operations and a final heat treatment in the final dimension, characterised in that the inner component of zirconium or a zirconium alloy during the manufacture, from the production of an ingot up to the completion of a cladding tube, comprising forging, rolling, extrusion, heat treatment and final heat treatment, is only subjected to heat influence at temperatures in the alpha-phase range below the temperature when an incipient beta-phase transformation takes place.

Dwg.0/1

Abstract (Equivalent): US 5620536 A

A method of manufacturing a composite cladding tube of a nuclear fuel element which is resistant to pellet-clad interaction, the composite cladding tube comprising an inner portion formed from a first component selected from the group consisting of zirconium and a zirconium alloy and an outer portion formed from a second component selected from the group consisting of Zircaloy 2, Zircaloy 4 and Zr 2.5 Nb, the method comprising the steps of: (a) providing an ingot of the first component, (b) forging rolling, extruding, and heat treating the ingot of the first component to form an inner billet, (c) positioning the inner billet from step (b) within an outer machined billet formed from an ingot of the second component, (d) extruding the joined billets from step (c) to form a joined tube blank, and (e) machining the tube blank from step (d) to provide the composite cladding tube, (f) the steps (b)-(e) being conducted at a temperature below that which causes incipient beta-phase transformation within the first component.

Dwg.0/1

Title Terms: MANUFACTURE; NUCLEAR; FUEL; ELEMENT; FUEL; ROD; CLAD; TUBE; ZIRCONIUM; CLAD; TUBE; ZIRCONIUM; ALLOY; INTERNAL; LINING; TEMPERATURE; MANUFACTURE; TEMPERATURE; INCIPIENT; PHASE; TRANSFORM; BETA; PHASE; PLACE  
Derwent Class: K05; M21; M26  
International Patent Class (Main): G21C-000/00; G21C-003/20; G21C-021/02  
International Patent Class (Additional): C21D-009/08; C22F-001/00; C22F-001/18; G21C-003/06; G21C-003/07

35/7,DE/32 (Item 7 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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009375448

WPI Acc No: 1993-068926/199309

Annealing of zirconium@ based alloy to improve nodular corrosion resistance - giving prod. used in the cladding of fuel elements in nuclear reactors

Patent Assignee: GENERAL ELECTRIC CO (GENE )

Inventor: TAYLOR D F

Number of Countries: 010 Number of Patents: 005

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
EP 529907	A1	19930303	EP 92307522	A	19920818	199309 B
US 5188676	A	19930223	US 91749052	A	19910823	199310

TW 198733	A	19930121	TW 92101397	A	19920225	199326
JP 5209257	A	19930820	JP 92220556	A	19920820	199338
JP 2677933	B2	19971117	JP 92220556	A	19920820	199751

Priority Applications (No Type Date): US 91749052 A 19910823

Cited Patents: GB 2086945; US 4000013; US 4351678

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 529907	A1	E		C22F-001/02	
Designated States (Regional): CH DE ES FR IT LI SE					
US 5188676	A		7	C22F-001/00	
JP 5209257	A		6	C22F-001/18	
JP 2677933	B2		6	C22F-001/18	Previous Publ. patent JP 5209257
TW 198733	A			C21D-001/26	

Abstract (Basic): EP 529907 A

A process for annealing a zirconium based alloy member having a cold worked or beta quenched crystal structure comprises: annealing the member in an atmos. consisting of oxygen and the balance an inert atmos. to form an adherent black oxide on the member. Also claimed is a process for recrystallisation annealing a zirconium based alloy as above.

Pref. the atmos. comprises at least 0.1 vol.% of oxygen, 0.1 grams of oxygen per square metre surface area of zirconium based alloy, less than 20 ppm of nitrogen, less than 2 ppm of hydrogen and less than 10 ppm of water.

USE/ADVANTAGE - Used in the cladding of fuel elements in nuclear reactors. The reductions in nodular corrosion resistance found in the annealed zirconium alloy member is maintained or improved.

Dwg.1/5

Abstract (Equivalent): US 5188676 A

Annealing zircalloy part comprises annealing to form an adherent black oxide, in atmos. at effective amt. of oxygen to form the black oxide, and the remainder an inert atmos.

Recrystallising annealing the zircalloy part comprises formation of adherent black oxide, in atmos. of O<sub>2</sub> and inert gases.

Atmos. e.g. comprises less than 20 pts. per million of N<sub>2</sub>, less than 2 pts. per million H<sub>2</sub> and less than 10 ppm water.

USE/ADVANTAGE - To anneal zircalloy part with cold worked or beta quenched crystal structure, to mitigate redn. modular corrosion resistance due to recrystallisation annealing.

Dwg.0/5

Title Terms: ANNEAL; ZIRCONIUM; BASED; ALLOY; IMPROVE; NODULE; CORROSION; RESISTANCE; PRODUCT; CLAD; FUEL; ELEMENT; NUCLEAR; REACTOR

Derwent Class: K05; M29; X14

International Patent Class (Main): C21D-001/26; C22F-001/00; C22F-001/02; C22F-001/18

International Patent Class (Additional): C22C-016/00

35/7,DE/33 (Item 8 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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009142826

WPI Acc No: 1992-270264/199233

Nuclear fuel element component of zirconium@ alloy - with high temp.  
corrosion resistance to cooling water, fuel and fission products

Patent Assignee: SIEMENS AG (SIEI )

Inventor: STEINBERG E

Number of Countries: 006 Number of Patents: 007

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week	
EP 498259	A2	19920812	EP 92101295	A	19920127	199233	B
US 5245645	A	19930914	US 91745904	A	19910816	199338	
			US 92839629	A	19920221		
EP 498259	A3	19930616	EP 92101295	A	19920127	199405	
US 5296058	A	19940322	US 91745904	A	19910816	199411	
			US 92839629	A	19920221		
			US 9338671	A	19930326		
EP 498259	B1	19960327	EP 92101295	A	19920127	199617	
DE 59205799	G	19960502	DE 505799	A	19920127	199623	
			EP 92101295	A	19920127		
ES 2084847	T3	19960516	EP 92101295	A	19920127	199627	

Priority Applications (No Type Date): EP 91112979 A 19910801; EP 91101463 A 19910204

Cited Patents: No-SR.Pub; EP 171675; EP 196286; EP 405172; EP 85553; FR 1327734; FR 2509510; US 4963323

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 498259	A2	G	10	G21C-003/07	
Designated States (Regional): DE ES FR GB SE					
US 5245645	A		9	G21C-003/06	CIP of application US 91745904
US 5296058	A		10	C22C-016/00	Cont of application US 91745904
					Div ex application US 92839629
					Div ex patent US 5245645
EP 498259	B1	G	12	G21C-003/07	
Designated States (Regional): DE ES FR GB SE					
DE 59205799	G			G21C-003/07	Based on patent EP 498259
ES 2084847	T3			G21C-003/07	Based on patent EP 498259
EP 498259	A3			G21C-003/07	

Abstract (Basic): EP 498259 A

A structural part for a nuclear fuel element, esp. a cladding tube or a spacer, consists of a Zr alloy contg. (by wt.) 0.8-1.7 % Sn, at least one of 0.07-0.5% Fe, 0.05-0.35% Cr and up to 0.1% Ni, 700-2000 ppm. O and/or up to 150 ppm. Si, balance Zr and impurities. Secondary phases of Fe, Cr and Ni pptd. in the matrix have a geometrical mean diameter of 0.1-0.3 microns and the alloy is in the max. 10% recrystallised state, a specimen of the alloy having an average grain size of max. 3 microns when recrystallisation annealed to a 95-99% recrystallised state.

Also claimed are (i) processes for producing the structural parts; and (ii) structural parts in which the alloy is Zircaloy-2 or -4.

ADVANTAGE - The structural part has high corrosion resistance w.r.t. the fuel and fission products and w.r.t. cooling water even when used at relatively high temp., e.g. the temp. which prevails in a PWR and which is higher than that in a BWR

Dwg.0/3

Abstract (Equivalent): EP 498259 B

A structural part for a nuclear fuel element, esp. a cladding tube or a spacer, consists of a Zr alloy contg. (by wt.) 0.8-1.7 % Sn, at least one of 0.07-0.5% Fe, 0.05-0.35% Cr and up to 0.1% Ni, 700-2000 ppm. O and/or up to 150 ppm. Si, balance Zr and impurities. Secondary phases of Fe, Cr and Ni pptd. in the matrix have a geometrical mean diameter of 0.1-0.3 microns and the alloy is in the max. 10% recrystallised state, a specimen of the alloy having an average grain size of max. 3 microns when recrystallisation annealed to a 95-99% recrystallised state.

Also claimed are (i) processes for producing the structural parts;

and (ii) structural parts in which the alloy is Zircaloy-2 or -4.

ADVANTAGE - The structural part has high corrosion resistance w.r.t. the fuel and fission products and w.r.t. cooling water even when used at relatively high temp., e.g. the temp. which prevails in a PWR and which is higher than that in a BWR

EP-498259 Structural part for a reactor fuel element, in particular a cladding tube for a fuel rod filled with nuclear fuel or spacers for such fuel rods, having the following features: (a) the material of the structural part is a zirconium alloy with at least one alloy component of the group oxygen and silicon, with tin as alloy component, having at least one alloy component of the group iron, chromium and nickel and with a remainder consisting of zirconium and unavoidable contaminants, (b) in the zirconium alloy there is selected a content of oxygen in the range of 700 to 2000 ppm, a content of silicon of up to 150 ppm, a content of iron in the range of 0.07 to 0.5% by weight, a content of chromium in the range of 0.05 to 0.35% by weight, a content of nickel of up to 0.1% by weight and a content of tin in the range of 0.8 to 1.7% by weight, (c) the geometric mean value of the diameter of the alloy components of the group iron, chromium and nickel, precipitated in the matrix of the zirconium alloy as secondary phases, is selected in the range of 0.1 to 0.3 micro-m, and (d) the degree of recrystallisation of the zirconium alloy is less than or equal to 10% and a sample of the zirconium alloy has after a recrystallisation annealing with a degree of recrystallisation of 97+/-2% a geometric mean value of the grain diameter which is smaller than or equal to 3 micro-m.

(Dwg.0/3)

Abstract (Equivalent): US 5245645 A

The structural part comprises a Zr alloy contg. 0.07-0.5 wt.% Fe, 0.05-0.35 wt.% Cr, upto 0.1 wt.% Ni and 0.8-1.7 wt.% Sn, with 700-2000 ppm O and upto 150 ppm Si. The Fe, Cr and Ni are pptd. out of a matrix of the Zr alloy as sec. phases, having a dia. with a geometric average value of 0.1-0.3 microns. The deg. of recrystallisation of the Zr alloy is upto 10%. A sample of the alloy after a recrystallisation annealing has a deg. of recrystallisation of 97 +/- 2% and has a grain size with a geometric mean value upto 3 microns. The alloy pref. has a texture with a Kearns parameter  $fr = 0.6$  to  $0.8$ . Ratio of Fe to Cr is pref. 2:1.

USE/ADVANTAGE - Used for a cladding or casing tube for a fuel rod or spacer. High corrosion resistance both to the nuclear fuel and fission prods. but also to the water coolant, even at relatively high temps. (Dwg.0/3)

US5296058 The structural part is produced by (d) annealing a Zr alloy in the beta range below the m.pt to dissolve pptd. out alloy ingredients, then quenching at least 30 deg.C./s, at a temp. transition through the alpha+beta range in which both hexagonal and bcc structures are present, (b) annealing at a 1st temp. in the alpha range until formation of pptes. of the alloy ingredients having a ppte. dia. with a geometric mean value of 0.1-0.3 microns, (c) hot forging at a 2nd temp. in the alpha range below the 1st temp., (d) hot rolling or hot extending at a temp. in the alpha range below the 1st. temp., and (e) cold rolling in at least two steps, including recrystallisation annealing between the rolling steps with a recrystallisation deg. of 95-99% at a temp. in the alpha range, while cold pilgering in at least two steps sepd. by recrystallisation annealing.

USE/ADVANTAGE - Used as cladding or casing for a nuclear fuel rod or spacer. Optimal corrosion resistance to water at elevated temps.

(Dwg.0/3)

Title Terms: NUCLEAR; FUEL; ELEMENT; COMPONENT; ZIRCONIUM; ALLOY; HIGH; TEMPERATURE; CORROSION; RESISTANCE; COOLING; WATER; FUEL; FISSION;

## PRODUCT

Derwent Class: K05; M26

International Patent Class (Main): C22C-016/00; G21C-003/06; G21C-003/07

International Patent Class (Additional): C22F-001/18

35/7,DE/34 (Item 9 from file: 351)

DIALOG(R)File 351:Derwent WPI

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004776906

WPI Acc No: 1986-280247/198643

Thin zirconium-niobium alloy tubing prodn. - useful for nuclear fuel cladding

Patent Assignee: WESTINGHOUSE ELECTRIC CORP (WESE )

Inventor: MCDONALD S G; SABOL G P

Number of Countries: 007 Number of Patents: 005

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
EP 198570	A	19861022	EP 86300259	A	19860116	198643 B
JP 61210166	A	19860918	JP 8611802	A	19860122	198644
ES 8708021	A	19871116	ES 551049	A	19860120	198751
EP 198570	B	19900829				199035
KR 9309986	B1	19931013	KR 86376	A	19860122	199438

Priority Applications (No Type Date): US 85693546 A 19850122

Cited Patents: A3...8741; EP 71193; EP 85553; FR 2576322; LU 41401;

No-SR.Pub; US 2894866; US 3341373; US 3865635

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 198570	A	E	19		
Designated States (Regional): BE FR GB IT					
EP 198570	B				
Designated States (Regional): BE FR GB IT					
KR 9309986	B1			C22F-001/18	

Abstract (Basic): EP 198570 B

Thin walled tubing is made from a Zr-Nb alloy, contg. 1-2.5 wt.% Nb as homogeneously dispersed finely divided particles, by (i) beta-treating a billet of the alloy, (ii) extruding at max. 650 deg.C to form a tube shell, (iii) multistage cold working with intermediate anneals at below 650 deg.C, and (iv) final annealing at below 600 deg.C. The resulting tubing has a microstructure contg. a homogeneous dispersion of Nb particles of less than 800 angstroms size.

USE/ADVANTAGE - The process is useful for prodn. of nuclear fuel cladding. Tubing, of 0.040 in. or less wall thickness and excellent corrosion resistance, is made without requiring extensively long final annealing times. (19pp Dwg.No.0/5)

Abstract (Equivalent): EP 198570 B

A process for fabricating thin-walled tubing having a wall thickness of about 1 mm or less from a zirconium-niobium alloy containing from 1 to 2.5 per cent by weight niobium as homogeneously dispersed finely divided particles and optionally up to 0.5 per cent by weight of copper, iron, molybdenum, nickel, tungsten, vanadium or chromium as a third element, balance, apart from impurities, zirconium, characterised by beta-treating a zirconium niobium alloy billet containing from 1 to 2.5 per cent by weight niobium; extruding said beta-treated billet at a temperature no higher than 650 deg C to form a tube shell; further deforming said tube shell by cold working the same in a plurality of cold working stages; annealing said tube shell,

between each of said stages of cold working, at a temperature below 650 deg C, and final annealing the resultant tubing at a temperature below 600 deg C, so as to produce a microstructure of the material having niobium particles of a size below about 800 angstroms (80nm) homogeneously dispersed therein.

Title Terms: THIN; ZIRCONIUM; NIOBIUM; ALLOY; TUBE; PRODUCE; USEFUL; NUCLEAR; FUEL; CLAD

Derwent Class: K05; M26; M29; P51

International Patent Class (Main): C22F-001/18

International Patent Class (Additional): B21C-023/01; B21C-029/00; B21C-037/06

35/7,DE/35 (Item 10 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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004756367

WPI Acc No: 1986-259708/198640

Mfr. of nuclear fuel cladding tubes of zirconium alloy - using annealing temp. and time sufficient to equilibrate pptd. sec. phase particles to improve corrosion resistance

Patent Assignee: SANTRADE LTD (SANV )

Inventor: ANDERSSON E T; SCHEMEL J H; WILSON S A

Number of Countries: 008 Number of Patents: 009

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
EP 196286	A	19861001	EP 86850081	A	19860306	198640 B
SE 8501216	A	19860913				198645
SE 8501217	A	19860913				198645
JP 61270360	A	19861129	JP 8651584	A	19860311	198702
EP 196286	B	19890517				198920
DE 3663372	G	19890622				198926
US 4908071	A	19900313	US 88226517	A	19880729	199016
KR 9309987	B1	19931013	KR 861739	A	19860311	199438
JP 2583488	B2	19970219	JP 8651584	A	19860311	199712

Priority Applications (No Type Date): SE 851217 A 19850320; SE 851216 A 19850320

Cited Patents: FR 2219978; GB 2079317; US 4450016

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 196286	A	E	6	P	
Designated States (Regional): DE FR GB IT					
EP 196286	B	E		P	
Designated States (Regional): DE FR GB IT					
JP 2583488	B2		5	P	Previous Publ. patent JP 61270360
SE 8501216	A			P	
SE 8501217	A			P	
JP 61270360	A			P	
DE 3663372	G			P	
US 4908071	A			P	
KR 9309987	B1			P	

Abstract (Basic): EP 196286 A

Mfr. nuclear fuel cladding tubes of zirconium alloy contg. 1-5 wt.% alloying elements such as Sn, Fe, Cr and Ni to improve the corrosion resistance in a water-cooled reactor environment is claimed. Method comprises annealing the material, after extrusion and/or between cold rollings, at 625-790 deg.C in the alpha phase range and at a



combination of temp. and time which gives complete equilibrium between Zr matrix and the precipitated secondary phase particles. An annealing parameter (A) is used, defined by formula (I) where  $t$  = annealing time (hours),  $R$  = gas constant (cal/mole.degree),  $T$  = temp. (K), and 65000 is the activation energy (cal/mole).

The value of this parameter (A) is kept above a critical value of  $2.3 \times 10$  power -14, so that min. annealing times at various temps. are as follows: 0.5 h at 790 deg.C; 3.9 h at 725 deg.C; 22.2 h at 675 deg.C; 56.5 h at 650 deg.C; and 151.3 h at 625 deg.C.

ADVANTAGE - Equilibration of the precipitated particles ensures a minimum dissolved Fe content in the Zr matrix, thus improving corrosion resistance.

Dwg.0/0

EP 196286 B

Mfr. nuclear fuel cladding tubes of zirconium alloy contg. 1-5 wt.% alloying elements such as Sn, Fe, Cr and Ni to improve the corrosion resistance in a water-cooled reactor environment is claimed. Method comprises annealing the material, after extrusion and/or between cold rollings, at 625-790 deg.C in the alpha phase range and at a combination of temp. and time which gives complete equilibrium between Zr matrix and the precipitated secondary phase particles. An annealing parameter (A) is used, defined by formula (I) where  $t$  = annealing time (hours),  $R$  = gas constant (cal/mole.degree),  $T$  = temp. (K), and 65000 is the activation energy (cal/mole).

The value of this parameter (A) is kept above a critical value of  $2.3 \times 10$  power -14, so that min. annealing times at various temps. are as follows: 0.5 h at 790 deg.C; 3.9 h at 725 deg.C; 22.2 h at 675 deg.C; 56.5 h at 650 deg.C; and 151.3 h at 625 deg.C.

ADVANTAGE - Equilibration of the precipitated particles ensures a minimum dissolved Fe content in the Zr matrix, thus improving corrosion resistance. (6pp Dwg.No.0/0)

Abstract (Equivalent): EP 196286 B

Method of making cladding tubes of a zirconium alloy containing 1-5 percent by weight of alloying elements including S, Fe, Cr and Ni and the rest essentially Zr, for the purpose of improving the corrosion resistance to general corrosion in media typical of water cooled thermal nuclear reactors at high pressure and high temperature, at which the material is annealed after extrusion and/or between cold rollings in the alpha-phase range at a temperature, within the interval 625-790 deg.C, characterised in that there is used a combination of temperature and time which essentially gives complete equilibrium between zirconium matrix and the precipitated secondary phase particles at which the combination of temperature/time is defined by an annealing parameter where  $t$  is the annealing time in hours,  $T$  is the absolute temperature,  $R$  is the general gas constant in cal/mole - degree, which at equilibrium shall exceed a critical value  $A_c = 2.3, 10^{-14}$ , where said avlue means the following examples of shortest annealing times within the temperature interval 625-790 deg.C. Annealing temp. (deg.C), annealing time (hours) 790, 0.5; 725, 3.9; 675, 22.2; 650, 56.5; 625, 151.3.

Abstract (Equivalent): US 4908071 A

Cladding tubes of Zr alloy contg. 1-5 wt.% alloying elements including Sn, Fe, Cr and Ni, for use in nuclear reactors, are mfd. from extruded material which is annealed after extrusion and between cold rollings in the alpha phase range at 625-790 deg.C. Combination of temp. and time gives equilibrium between Zr matrix and pptd. second phase particles. Annealing temp. and time are defined by a parameter  $A = t.e \text{ power } (-6500/RT)$ , where  $t$  is annealing time,  $T$  is absolute temp., and  $R$  is the gas constant. Tubes are cooled at 3 deg.C/min. max.

ADVANTAGE - Improved corrosion resistance for longer service times.

(4pp)

Title Terms: MANUFACTURE; NUCLEAR; FUEL; CLAD; TUBE; ZIRCONIUM; ALLOY;  
ANNEAL; TEMPERATURE; TIME; SUFFICIENT; EQUILIBRIUM; PRECIPITATION; SEC;  
PHASE; PARTICLE; IMPROVE; CORROSION; RESISTANCE

Derwent Class: K05; M29

International Patent Class (Main): C22F-001/18

International Patent Class (Additional): G21C-003/06; G21C-003/20

35/7,DE/36 (Item 11 from file: 351)

DIALOG(R)File 351:Derwent WPI

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004722285

WPI Acc No: 1986-225627/198635

Zirconium(alloy) or titanium (alloy) seamless tube mfr. - by  
thermo-mechanical treatment of welded tube

Patent Assignee: WESTINGHOUSE ELECTRIC CORP (WESE )

Inventor: BARRY R F; SABOL G P

Number of Countries: 004 Number of Patents: 004

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
BE 904221	A	19860812	BE 904221	A	19860212	198635 B
JP 61186462	A	19860820	JP 8628067	A	19860213	198640
FR 2580524	A	19861024	FR 861951	A	19860213	198649
US 4690716	A	19870901	US 85701326	A	19850213	198737

Priority Applications (No Type Date): US 85701326 A 19850213

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
BE 904221	A		14		

Abstract (Basic): BE 904221 A

Seamless tube is mfd. from a welded starting tube of zirconium (alloy) or titanium (alloy), having a heterogeneous structure as a result of welding, by (i) through-heating of successive axial segments of the welded tube, including the weld, for uniform transformation to the beta phase and (ii) quenching of the tube segments, steps (i) and (ii) being affected rapidly to avoid beta phase grain growth beyond 200 microns diameter. Finally (iii) the cooled tube is rapidly deformed to final shape.

USE/ADVANTAGE - Resulting tubes are useful for nuclear fuel cans, aviation hydraulic piping, heat exchanger tubes, and condenser tubes. It has a uniform alpha phase structure with improved mechanical properties and corrosion resistance. (14pp Dwg.No.0/0)

Abstract (Equivalent): US 4690716 A

Seamless tubes of zirconium or titanium alloy are formed by welding together a precursor tube to give a heterogeneous phase.

The tube is then heated in successive stages to convert all the material, including the weld zone, into beta phase material before quenching sufficiently rapidly to give a fine grained beta structure with no grain above 200 microns. Finally the quenched tube is subject to cold reduction until the final size and shape is reached.

ADVANTAGE - The process produces seamless tube suitable e.g. as nuclear fuel rod cladding, at reduced cost. (4pp)

Title Terms: ZIRCONIUM; ALLOY; TITANIUM; ALLOY; SEAM; TUBE; MANUFACTURE;  
THERMO; MECHANICAL; TREAT; WELD; TUBE

Derwent Class: K05; M21; P51

International Patent Class (Additional): B21B-003/02; B21B-021/00;

C21B-000/00; C21D-008/10; C22F-001/18

35/7,DE/37 (Item 12 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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003740419

WPI Acc No: 1983-736618/198333

Fabricating zirconium alloys to improve steam corrosion resistance - by  
reducing temp. of working and annealing during fabrication

Patent Assignee: WESTINGHOUSE ELECTRIC CORP (WESE )

Inventor: MCDONALD S G; SABOL G P

Number of Countries: 009 Number of Patents: 007

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
EP 85553	A	19830810	EP 83300455	A	19830128	198333 B
JP 58204144	A	19831128	JP 8313618	A	19830128	198402
ES 8602148	A	19860301	ES 519378	A	19830128	198619
US 4584030	A	19860422	US 84571122	A	19840113	198619
CA 1214978	A	19861209				198702
EP 85553	B	19881123				198847
DE 3378537	G	19881229				198902

Priority Applications (No Type Date): US 82343787 A 19820129; US 84571122 A 19840113

Cited Patents: BE 691169; FR 1415082; No-SR.Pub; US 3431104; US 3567522; US 4000013; US 3645800

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
EP 85553	A	E	19		

Designated States (Regional): DE FR GB IT SE

EP 85553	B	E
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Designated States (Regional): DE FR GB IT SE

Abstract (Basic): EP 85553 A

Zircalloy article has a microstructure adjacent the surface which comprises a random distribution of precipitates (below 1100 microns) and the surface after 5 days exposure to 454 deg.C, 10.3 MPa steam has an adherent oxide film. The Zircalloy article is fabricated by conventional beta treatment of a billet followed by thermomechanical working below 625 deg.C. The working may be cold working with intermediate anneals at 500-600 deg.C. Used as cladding for nuclear components in pressurised water and boiling water reactors. Improved high temp. steam corrosion resistance is obtd.

Abstract (Equivalent): EP 85553 B

A process for fabricating Zircalloy alloy shapes comprising the steps of heating a Zircalloy intermediate product to an elevated temperature above the alpha+beta to beta transus temperature and quenching said Zircalloy intermediate product from said elevated temperature to a temperature below the alpha+beta to alpha transus temperature to form precipitates having an average diameter below about 1100 Angstroms, then extruding said alloy at a temperature between about 500 and 600 deg.C, then cold working said alloy in series of cold pilgering steps each of which is preceded by a thermal treatment step comprising essentially of only a low temperature anneal, said low temperature anneal limited to about 500 to 600 deg.C, and after the final cold pilgering step subjecting the resulting material to a final anneal at about 466 to 600 deg.C. (11pp)

Abstract (Equivalent): US 4584030 A

Zircalloy workpiece has a microstructure region adjacent to a

surface which comprises a random distribution of pptes. The pptes have an average size of less than 1100 angstroms and the surface is exposed for 5 days to steam at a pressure of 10.3 MPa and temp of 454 deg.C to form a uniform, continuous, adherent oxide film.

Pref. the microstructure comprises a stress relieved cold worked structure, pref comprising polygonal alpha grains and an anisotropic crystallographic structure.

ADVANTAGE - Good long term corrosion resistances in a high temp steam environment. (11pp)

Title Terms: FABRICATE; ZIRCONIUM; ALLOY; IMPROVE; STEAM; CORROSION; RESISTANCE; REDUCE; TEMPERATURE; WORK; ANNEAL; FABRICATE

Derwent Class: K05; M29; X14

International Patent Class (Additional): C22C-016/00; C22F-001/18; G21C-003/06

35/7,DE/38 (Item 13 from file: 351)

DIALOG(R)File 351:Derwent WPI

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003715170

WPI Acc No: 1983-711353/198328

Nuclear fuel element cladding mfr. - by cold working and heat treating zirconium alloy tube lined with zirconium to recrystallise the latter

Patent Assignee: GENERAL ELECTRIC CO (GENE )

Inventor: DAVIES J H; ROSENBAUM U S

Number of Countries: 001 Number of Patents: 001

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
US 4390497	A	19830628				198328 B

Priority Applications (No Type Date): US 81304011 A 19810921; US 7945225 A 19790604

Patent Details:

Patent No	Kind	Lan	Pg	Main IPC	Filing Notes
US 4390497	A		8		

Abstract (Basic): US 4390497 A

Method is claimed for making nuclear fuel element cladding tube from a tube shell consisting of Zr alloy contg. more than 5000 ppm impurities, with a protective barrier bonded to the inside of the tube shell and consisting of Zr contg. less than 5000 ppm impurities with thickness 1-30% of that of the composite tube.

The method comprises (i) reducing the dia. of the tube shell and lining layer by cold working to obtain the desired dia. and wall thickness; (ii) heat treating between each reduction step to fully recrystallise the Zr alloy; and (iii) finally heat treating to complete recrystallisation of the Zr metal layer to produce a fine-grained microstructure and stress-relieves but does not fully recrystallise the Zr alloy.

The heat treatment enables the Zr metal layer to be recrystallised without causing undesirable grain growth. The microstructure of the protective barrier layer can be further improved by shot peening. The Zr alloy retains elongated grain structure and has higher strength at high strain rates.

Title Terms: NUCLEAR; FUEL; ELEMENT; CLAD; MANUFACTURE; COLD; WORK; HEAT; TREAT; ZIRCONIUM; ALLOY; TUBE; LINING; ZIRCONIUM; RECRYSTALLISATION; LATTER

Derwent Class: K05; M29; X14

International Patent Class (Additional): G21C-003/20

35/7,DE/39 (Item 14 from file: 351)  
DIALOG(R)File 351:Derwent WPI  
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003425644

WPI Acc No: 1982-00021J/198247

Zirconium alloy nuclear fuel cladding tube prodn. - using controlled  
intermediate anneals during cold rolling

Patent Assignee: ASEA ATOM AB (ALLM )

Number of Countries: 006 Number of Patents: 006

Patent Family:

Patent No	Kind	Date	Applicat No	Kind	Date	Week
BE 893788	A	19821103				198247 B
DE 3224685	A	19830120				198304
SE 8104214	A	19830207				198308
FR 2509510	A	19830114				198309
JP 58025466	A	19830215				198312
FI 8202395	A	19830228				198315

Priority Applications (No Type Date): SE 814214 A 19810707

Patent Details:

Patent No	Kind	Lan Pg	Main IPC	Filing Notes
BE 893788	A	13		

Abstract (Basic): BE 893788 A

Zirconium alloy nuclear fuel cladding tubes are produced by (i) extrusion at below 680 deg.C; (ii) cold rolling with intermediate anneals, at least one of which is carried out at above 650 deg.C in the alpha phase region and is followed by cooling at at least 5 deg.C/min. from the annealing temp. to 650 deg.C, the intermediate anneals after this at least one anneal being carried out at max. 600 deg.C; and (iii) final annealing.

The resulting tubes contain a homogeneous distribution of very fine second phase particles giving good nodular corrosion resistance and good mechanical properties.

Title Terms: ZIRCONIUM; ALLOY; NUCLEAR; FUEL; CLAD; TUBE; PRODUCE; CONTROL; INTERMEDIATE; ANNEAL; COLD; ROLL

Derwent Class: K05; M21; M29; P51; P54; X14

International Patent Class (Additional): B21C-023/00; B23B-000/00; C22C-016/00; C22F-001/00; G21C-003/06; G21C-021/00

35/7,DE/40 (Item 1 from file: 32)  
DIALOG(R)File 32:METADEX(R)  
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0905565 MA Number: 199209-61-0661

A Method of Manufacturing Cladding Tubes for Fuel Rods for Nuclear Reactors.

Andersson, T

Sandvik

Patent: EP0425465, European Patent 26 Oct. 1990

Auszuge aus den Europäischen Patentanmeldungen, Teil I 7, (18), 1722 2  
May 1991

Country of Publication: Germany

Journal Announcement: 9209

Document Type: Patent

Language: ENGLISH

Abstract: A method of making cladding tubes for fuel rods of a Zr alloy containing (in wt.%) 1.2-1.7 wt.% Sn, 0.07-0.24 Fe, 0.05-0.15 Cr, 0-0.08 Ni and the balance essentially Zr and ordinary impurities for the purpose of improving the resistance to nodular corrosion under operating conditions in boiling water nuclear reactors with or without a barrier of Zr having a Sn addition of 0-0.5 bonded to its inside surface includes the steps of: (a) extruding a perforated billet of the Zr-based alloy within the alpha -phase range for obtaining a tube, (b) cold rolling the tube with multiple cold working passes to the size for the final cold rolling and annealing the tube at a temperature in the alpha -phase range subsequent to each cold rolling pass, (c) heating an outer portion of the tube wall to the beta -phase range for a time sufficient to transform the outer portion of the tube wall to beta -phase while cooling the inner portion of the tube wall at a temperature sufficiently low that essentially no metallurgical changes occur at the inner portion of the tube wall and then cooling the tube sufficiently rapidly to transform the beta -phase into a structure of alpha -grains with a fine distribution of intermetallic particles in the alpha -grain boundaries and thereby improve the resistance to nodular corrosion; (d) cold rolling the thus partially beta -quenched tube to the final cladding tube size; (e) annealing the cold rolled tube at a temperature of 400-650 deg C.

Descriptors: Patent; Zirconium base alloys-- Claddings; Nuclear fuel elements; Tubemaking; Cold rolling; Heat treatment; Corrosion resistance-- Microstructural effects; Microstructure-- Cooling effects

WG.24  
EAST

S nuclear \$1 near 3 fuel near 3 clad \$3  
→ H1 + (Zircon: \$3 or Zircaloy \$3)  
→ lots of good art in Japan Derwent

DERWENT-ACC-NO: 1999-266204  
DERWENT-WEEK: 200051  
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TITLE: Zirconium alloy nuclear fuel cladding production

INVENTOR: ISOBE, T; SUDA, Y

PATENT-ASSIGNEE: MITSUBISHI MATERIALS CORP[MITV]

PRIORITY-DATA: 1998JP-0287800 (October 9, 1998) ,  
1997JP-0278935 (October 13,  
1997)

PATENT-FAMILY:

PUB-NO	PUB-DATE	LANGUAGE
PAGES	MAIN-IPC	
US 6125161 A	September 26, 2000	N/A
000	G21C 003/07	
FR 2769637 A1	April 16, 1999	N/A
039	C21D 008/00	
JP 11194189 A	July 21, 1999	N/A
024	G21C 003/06	

APPLICATION-DATA:

PUB-NO	APPL-DESCRIPTOR	APPL-NO
APPL-DATE		
US 6125161A	Div ex	1998US-0169968
	October 13, 1998	
US 6125161A	N/A	1999US-0397094
	September 16, 1999	
FR 2769637A1	N/A	1998FR-0012784
	October 13, 1998	
JP 11194189A	N/A	1998JP-0287800
	October 9, 1998	

INT-CL (IPC): C21D001/26; C21D008/00 ; C22C016/00 ;  
C22F001/00 ;  
C22F001/18 ; G21C003/06 ; G21C003/07

ABSTRACTED-PUB-NO: FR 2769637A

BASIC-ABSTRACT: NOVELTY - In the production of nuclear fuel  
cladding of a  
zirconium alloy containing Nb or Nb+Ta, annealing is  
carried out at 550-850

deg. C for 1-4 h such that the log of the cumulative anneal parameter is -20 to -15 and satisfies a mathematical relationship relating it to the Nb or Nb+Ta content.

DETAILED DESCRIPTION - Nuclear fuel cladding is produced by subjecting a zirconium alloy of composition (by wt.) 0.2-1.7% Sn, 0.18-0.6% Fe, 0.07-0.4% Cr, 0.05-1.0% Nb, optionally 0.01-0.1% Ta, balance zirconium and impurities, including at most 60 ppm N, to hot forging, solution heat treatment, hot extrusion, repeated annealing and cold rolling, and final stress relief annealing, the annealing being carried out at 550-850 deg. C for 1-4 h such that the cumulative anneal parameter approx.  $SA_i$  (where approx.  $SA_i = \log \text{ asterisk } \exp(-40000/T_i)$ ) satisfies the relationships of  $-20$  to  $-15$  and  $\log \text{ approx. } SA_i = -18 - 10X_{Nb}$  to  $-15 - 3.75(X_{Nb} - 0.2)$ , in which  $A_i =$  anneal parameter for the 'i'th anneal,  $t_i =$  anneal duration (h) for the 'i'th anneal,  $T_i =$  the anneal temperature (K) for the 'i'th anneal and  $X_{Nb} =$  the Nb and optional Ta content (in wt.%). An INDEPENDENT CLAIM is also included for a zirconium alloy nuclear fuel cladding made by the above process.

USE - For producing nuclear fuel cladding tubes for a PWR.

ADVANTAGE - The annealing conditions provide the cladding tube with better corrosion resistance and creep properties than conventional cladding tubes and thus has a long useful life.

ABSTRACTED-PUB-NO: US 6125161A  
EQUIVALENT-ABSTRACTS: NOVELTY - In the production of nuclear fuel cladding of a zirconium alloy containing Nb or Nb+Ta, annealing is carried out at 550-850 deg. C for 1-4 h such that the log of the cumulative



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to -15 and satisfies a mathematical relationship relating  
it to the Nb or Nb+Ta  
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DETAILED DESCRIPTION - Nuclear fuel cladding is produced by  
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zirconium alloy of composition (by wt.) 0.2-1.7% Sn,  
0.18-0.6% Fe, 0.07-0.4%  
Cr, 0.05-1.0% Nb, optionally 0.01-0.1% Ta, balance  
zirconium and impurities,  
including at most 60 ppm N, to hot forging, solution heat  
treatment, hot  
extrusion, repeated annealing and cold rolling, and final  
stress relief  
annealing, the annealing being carried out at 550-850 deg.  
C for 1-4 h such  
that the cumulative anneal parameter approx.  $SA_i$  (where  
approx.  $SA_i = \text{approx.}$   
 $St_i \text{ asterisk } \exp(-40000/T_i)$ ) satisfies the relationships of  
log approx.  $SA_i =$   
-20 to -15 and log approx.  $SA_i = -18-10X_{Nb}$  to  
-15-3.75( $X_{Nb}-0.2$ ), in which  $A_i =$   
anneal parameter for the 'i'th anneal,  $t_i =$  anneal duration  
(h) for the 'i'th  
anneal,  $T_i =$  the anneal temperature (K) for the 'i'th  
anneal and  $X_{Nb} =$  the Nb  
and optional Ta content (in wt.%). An INDEPENDENT CLAIM is  
also included for a  
zirconium alloy nuclear fuel cladding made by the above  
process.

USE - For producing nuclear fuel cladding tubes for a PWR.

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tube with better  
corrosion resistance and creep properties than conventional  
cladding tubes and  
thus has a long useful life.

CHOSEN-DRAWING: Dwg.0/0

TITLE-TERMS:  
ZIRCONIUM ALLOY NUCLEAR FUEL CLAD PRODUCE

DERWENT-CLASS: K05 M26 M29 X14

*Nuclear Reactors*

CPI-CODES: K05-B04B; M26-B06; M26-B06C; M26-B06J; M26-B06N;  
M26-B06T; M29-C01;

*non-ferrous alloys.*

EPI-CODES: X14-B04; X14-B04A;

SECONDARY-ACC-NO:

CPI Secondary Accession Numbers: C1999-078658

Non-CPI Secondary Accession Numbers: N1999-198505

> to search in Derwent

K05-B04B.cpi. or

K05\$.cpi. < truncate.

Harry,

I obtained the first results in this printout (L61 and L62) from searching the claim language and some process conditions. However, I was unhappy with those results, so I searched on Zr alloy AND nuclear fuel cladding AND manufacture/fabricate... AND pipe... I was happier with these results, they are listed at the end of the printout.

John

=> d his nofile

FILE 'REGISTRY' ENTERED AT 10:47:54 ON 12 JUL 2002

L1 55532 SEA ZR/ELS AND AYS/CI  
L2 53412 SEA ZR AND AYS/CI  
L3 45373 SEA L1 AND 50-100/MAC

FILE 'HCAPLUS' ENTERED AT 10:48:53 ON 12 JUL 2002

L4 1297849 SEA ALPHA?  
L5 59969 SEA L4(3A)(PHASE? OR STRUCTUR? OR MICROSTRUCTUR?)  
L6 178683 SEA MICROSTRUCTUR?  
L7 QUE PRODUC? OR PROD# OR GENERAT? OR MANUF? OR MFR# OR CREAT?  
OR FORM## OR FORMING# OR FORMAT? OR MAKE# OR MADE# OR MAKING#  
OR FABRICAT? OR PREPAR? OR PREP#  
L8 1084859 SEA CREEP? OR STRESS? OR STRAIN? OR DEFORMAT? OR FATIGUE? OR  
FRACTUR? OR EMBRITTLE?  
L9 176174 SEA L8(4A)(RESIST? OR RECOVER? OR STRENGTH?) OR TOUGHNESS? OR  
RESILEN?  
L10 737678 SEA NUCLEAR?  
L11 193290 SEA L10(4A)FUEL? OR URANIUM? OR PLUTONIUM OR PU  
L12 5494 SEA L10(3A)FUEL?(3A)CLAD?  
L13 QUE CLAD? OR CASING? OR SHEATH? OR ENSHEATH? OR ENCAS? OR  
ENCAPSULAT? OR ENVELOP? OR OVERLAID? OR LAMIN? OR LAMEL? OR  
ENCAS? OR WRAP? OR SURROUND?  
L14 695326 SEA LATH? OR STRIP? OR SWATH# OR BAND## OR SLAT### OR ROW###  
L15 694636 SEA TUBE# OR TUBING# OR TUBUL? OR TUBIFORM? OR PIPE# OR  
PIPING# OR PIPELI? OR CONDUIT? OR CYLIND?  
L16 27688 SEA (FAST## OR SWIFT## OR RAPID? OR QUICK?)(4A)(COOL?)  
L17 25940 SEA (COLD# OR METAL?)(2A)(WORK? OR METALWORK?)  
L18 250121 SEA ANNEAL? OR RECRYSTALLI?  
L19 19467 SEA ZIRCALOY? OR (ZIRONI## OR ZR)(3A)(ALLOY? OR AMALGAM? OR  
MIXTUR? OR ADMIX? OR BLEND?)  
L20 46961 SEA L3  
L21 57035 SEA L19 OR L20 OR ZIRCALOY(2W)4  
L22 7517 SEA ACICULAR  
L23 4269 SEA NEEDLE?(3A)(LIKE#)  
L24 11713 SEA L22 OR L23  
L25 QUE HEAT? OR WARM? OR HOT# OR CALEFACT? OR TORREFACT? OR  
PYROL? OR THERMOL? OR TEPEFACT? OR MELT? OR DISSOL? OR FUSE#  
OR FUSING# OR FUSION? OR (HIGH## OR HEIGHTEN OR ELEVAT?)(2A)(TE  
MP# OR TEMPERATUR?)  
  
L26 946440 SEA 70/SC,SX OR 71/SC,SX  
L27 599536 SEA 56/SX,SC

L28 QUE BETA  
L29 QUE (BINARY OR DUAL OR TWO) (3A) PHASE?  
L30 30738 SEA L21 AND L7  
L31 2577 SEA L30 AND L4  
L32 1144 SEA L30 AND L5  
L34 66 SEA L32 AND (L14 OR LATH?)  
L35 34 SEA L34 AND L8  
L36 7 SEA L34 AND L9  
L37 130 SEA L31 AND L14  
L38 56 SEA L37 AND L8  
L39 13 SEA L37 AND L9  
L40 1 SEA L39 AND L10  
L41 2802 SEA L21 AND L12  
L42 292 SEA L41 AND L4  
L43 120 SEA L41 AND L5  
L44 190 SEA L42 AND L7  
L45 71 SEA L43 AND L7  
L46 190 SEA L44 AND L4  
L47 71 SEA L45 AND L5  
L48 71 SEA L44 AND L5  
L49 71 SEA L45 AND L4  
L50 80 SEA L46 AND L8  
L51 4 SEA L46 AND L9  
L52 71 SEA L47 OR L48 OR L49  
L53 38 SEA L52 AND L8  
L54 4 SEA L52 AND L9  
L55 293 SEA L41 AND L18  
L56 54 SEA L55 AND L17  
L57 2 SEA L56 AND L16  
L58 60 SEA L55 AND L4  
L59 39 SEA L58 AND L28  
L60 31 SEA L59 AND L7  
L61 1 SEA L60 AND L14  
L62 2 SEA L60 AND L9  
L63 15 SEA L36 OR L40 OR L51 OR L54 OR L57 OR L61 OR L62  
L64 5 SEA L39 NOT L63  
L65 2951 SEA L30 AND L15  
L66 561 SEA L65 AND L12  
L67 42 SEA L66 AND L9  
L68 2 SEA L67 AND L4

=> d L63 1-15 cbib abs hitind hitrn

L63 ANSWER 1 OF 15 HCAPLUS COPYRIGHT 2002 ACS  
2002:94998 Document No. 136:174386 Degradation of the mechanical properties  
of **Zircaloy-4** due to hydrogen embrittlement.  
Bertolino, G.; Meyer, G.; Ipina, J. Perez (Centro Atomico Bariloche and  
Instituto Balseiro, Bariloche, Argent.). Journal of Alloys and Compounds,  
330-332, 408-413 (English) 2002. CODEN: JALCEU. ISSN: 0925-8388.  
Publisher: Elsevier Science S.A..

AB During nuclear reactor operation, the embrittlement of components  
**made of Zr-based alloys** is obsd. The degrdn.  
of their mech. properties is due to the combined effect of H absorption  
and the damage caused by n irradiation. The authors studied the influence of H  
content on the fracture **toughness** of a **Zircaloy-**  
**4 alloy**. Compact tension (CT) specimens were obtained from a  
hot-rolled, annealed and finally cold-rolled material. The obsd.  
**microstructure** consisted of **.alpha.-Zr** rounded grains  
with diams. of **.apprx.15 .mu.m**. Selection of the tested material was  
guided by the need to perform expts. on samples with a texture equiv. to

the cladding components of Candu-type nuclear reactors. The specimens were fatigue pre-cracked and H charged before testing. Two different reactions were performed. Specimens with a final H content ranging from 10 to 400 ppm were obtained by electrochem. charging and those with a final concn. of up to 2000 ppm were charged by absorption under a gaseous atm. In both cases, an homogeneous distribution of dissolved H and hydride phases was obtained. The dependence of the **toughness** on temp. and H content was measured on CT specimens. The anal. was performed in terms of J-integral and resistance curves.

CC 71-12 (Nuclear Technology)

ST **Zircaloy** mech property hydrogen embrittlement nuclear reactor

IT Nuclear reactors

(during nuclear reactor operation, embrittlement of components  
**made of Zr-based alloys** is obsd.)

IT Alloys, properties

RL: PRP (Properties)

(during nuclear reactor operation, embrittlement of components  
**made of Zr-based alloys** is obsd.)

IT Fracture **toughness**

(influence of H content on fracture **toughness** of a  
**Zircaloy-4** alloy.)

IT **Microstructure**

(**microstructure** consisted of **.alpha.-Zr** rounded  
grains)

IT **Nuclear fuel element cladding**

(samples with texture equiv. to fuel cladding components of Candu-type  
nuclear reactors.)

IT **11068-95-4**

RL: PRP (Properties)

(Degrn. of mech. properties of)

IT 1333-74-0, Hydrogen, processes

RL: PEP (Physical, engineering or chemical process); PROC (Process)

(embrittlement; degrdn. of mech. properties of **Zircaloy-**  
**4** due to hydrogen embrittlement)

IT **11068-95-4**

RL: PRP (Properties)

(Degrn. of mech. properties of)

L63 ANSWER 2 OF 15 HCAPLUS COPYRIGHT 2002 ACS

2000:478927 Document No. 133:153873 Influence of microstructure on fretting fatigue behavior of a near-alpha titanium alloy. Satoh, Toyochi (Jet Engine Division, 3rd Research Center, Technical Research & Development Institute, Japan Defense Agency, Tachikawa, 190-8533, Japan). ASTM Special Technical Publication, STP 1367(Fretting Fatigue), 295-307 (English) 2000. CODEN: ASTTA8. ISSN: 0066-0558. Publisher: ASTM.

AB To investigate the effect of microstructure on the fretting fatigue behavior of a near-.alpha. titanium alloy, fretting fatigue tests were carried out using DAT54, which is used in compressor blades and disks in aircraft gas turbine engines. Two kinds of microstructure in DAT54 were **prepd.** using different soln. heat treatment temps.: one is the equiaxed **.alpha. + .alpha. lath microstructure** and the other is the transformed **.beta. structure**. The plain and fretting **fatigue strengths** for the equiaxed **.alpha. + .alpha. lath microstructure** are higher than for the transformed **.beta.** structure. Fretting reduced **fatigue strengths** by a factor of three for both materials. Sensitivity to microstructure in fretting fatigue is relatively low compared with plain fatigue. Shot peening improved fretting fatigue life, because of lower tangential force between the specimen and the contact pad and because of residual stress in

compression induced by shot peening treatment.

CC 56-10 (Nonferrous Metals and Alloys)

IT 180254-61-9, DAT54

RL: PEP (Physical, engineering or chemical process); PRP (Properties); TEM (Technical or engineered material use); PROC (Process); USES (Uses)  
(effect of microstructure on fretting fatigue behavior of near-alpha titanium alloy)

IT 180254-61-9, DAT54

RL: PEP (Physical, engineering or chemical process); PRP (Properties); TEM (Technical or engineered material use); PROC (Process); USES (Uses)  
(effect of microstructure on fretting fatigue behavior of near-alpha titanium alloy)

L63 ANSWER 3 OF 15 HCAPLUS COPYRIGHT 2002 ACS

2000:242474 Document No. 133:10033 Influence of cladding microstructure on the low enthalpy failures in RIA simulation tests. Garde, A. M. (ABB Combustion Engineering Nuclear Fuel, Windsor, CT, 06095, USA). ASTM Special Technical Publication, STP 1354 (Zirconium in the Nuclear Industry: Twelfth International Symposium, 1998), 234-255 (English) 2000. CODEN: ASTTA8. ISSN: 0066-0558. Publisher: ASTM.

AB A review with 30 refs. Welding used during the **prepn.** of specimens for Reactivity Initiated Accident (RIA) simulation testing from fuel rods previously irradiated in reactors introduces the following 4 microstructural changes in the specimen: (a) **annealing** of irradiation damage, (b) a change from an **.alpha.-phase structure** of Zircaloy to a transformed **.beta.** structure in the cladding, (c) dissolution of the hydride rim **formed** under the oxide on the cladding tube outer surface during normal irradiation and possible radial-oriented hydride reprecipitation at the transformed **.beta.** platelet boundaries, and (d) reprecipitation of 2nd-phase particles previously dissolved due to radiation damage. A 5th factor, the change in the texture of Zircaloy, is also introduced due to the welding operation. The possible effect of these 5 changes on the specimen fracture **toughness** and failure enthalpy is evaluated. The published data on the mechanical properties of irradiated and unirradiated transformed **.beta.** structure of Zircaloy charged with H are reviewed to evaluate the impact of the 5 anticipated microstructural changes on the failure enthalpy. The available RIA simulation test fracture data with low-failure enthalpy are reviewed. The limited failure path information available appears to indicate that microstructural factors have contributed to the low-enthalpy failures. The applicability of the results from the low-enthalpy RIA test failures to Light Water Reactor (LWR) nuclear fuel should be based on the representativeness of the RIA specimen microstructure to that of the LWR fuel cladding.

CC 71-0 (Nuclear Technology)

Section cross-reference(s): 56

IT Fracture **toughness**

Light-water nuclear reactors

**Nuclear fuel element cladding**

Nuclear reactor accident

Simulation and Modeling, physicochemical

Structural phase transition

Welding of metals

(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

IT 11068-94-3, Zircaloy-2 11068-95-4,  
Zircaloy-4

RL: PRP (Properties)

(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

IT 11068-94-3, Zircaloy-2 11068-95-4,  
Zircaloy-4

RL: PRP (Properties)

(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

L63 ANSWER 4 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1998:246006 Document No. 128:311490 Micromechanisms of fatigue crack propagation in Ti3Al based alloys. Wu, X.; Bowen, P. (School of Metallurgy and Materials, The University of Birmingham, UK). Materials Science and Technology, 14(3), 206-216 (English) 1998. CODEN: MSCTEP. ISSN: 0267-0836. Publisher: Institute of Materials.

AB Direct monitoring of the influence of the **.alpha.2** and **.beta.** phases on a growing crack was performed using an FEG SEM for Ti-23Al-9Nb-2Mo-1Zr-1.2Si (at.-%) and Ti-23Al-11Nb-0.9Si (at.-%) Ti3Al based alloys. Crack growth rates are faster across individual **.alpha.2 laths** than across **.beta. laths** and/or along **.alpha.2/.beta. lath** interfaces. Fatigue cracks propagate incrementally through the **.alpha.2 phase** by decohesion of a favored slip band rather than crossing it catastrophically in one cycle. The formation of intersecting slip bands can lead to a tortuous crack path and a decreased av. crack growth rate in the **.alpha.2 phase**. When a crack meets the **.beta.** phase, the most common phenomenon obsd. is crack deflection. The fatigue crack then extends continuously along **.alpha.2/.beta.** interfaces under the effect of a mixed mode local stress intensity factor range. The basket weave microstructure achieves the max. fatigue crack growth resistance from **.alpha.2/.beta.** interfaces. Bridging and blunting can reduce fatigue crack growth rate remarkably. However, crack bridging happens only with larger **.beta. laths** and blunting is mainly seen only for secondary cracks. The efficiency of bridging and blunting thus depends on the ratio of load bearing capability of the **.beta. laths** involved over the local effective **.DELTA.K** range. Mechanisms operating during fatigue crack propagation are also compared with those obsd. during monotonic fracture.

CC 56-12 (Nonferrous Metals and Alloys)

IT 120474-22-8, Aluminum 23 niobium 11 silicon 0.9 titanium 65.1 atomic  
206645-94-5, Aluminum 23 molybdenum 2 niobium 9 silicon 1.2  
titanium 63.8 zirconium 1 atomic

RL: PRP (Properties)

(fatigue crack propagation in Ti3Al based alloys)

IT 206645-94-5, Aluminum 23 molybdenum 2 niobium 9 silicon 1.2  
titanium 63.8 zirconium 1 atomic

RL: PRP (Properties)

(fatigue crack propagation in Ti3Al based alloys)

L63 ANSWER 5 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1997:127704 Document No. 126:189567 Damage maps of titanium alloys. Helbert, A. L.; Feaugas, X.; Clavel, M. (Div. Mecanique, Univ. Technol. Compiegne, Fr.). Revue de Metallurgie/Cahiers d'Informations Techniques, 93(12), 1539-1549 (French) 1996. CODEN: CITMDA. ISSN: 0035-1563. Publisher: Revue de Metallurgie.

AB Damage evolution was studied on four **.alpha./beta.** titanium alloys using the local approach to fracture. Tensile tests on notched and smooth specimens have been performed to fracture or interrupted before fracture. Besides, finite element calcns. were conducted for each specimen design to provide the mech. parameters in the bulk of the specimen during loading up to fracture. In order to quantify damage, midsections of the specimens were polished and etched. In each surface element, void d. and length were quantified and linked to the mech. parameters. Void nucleation has

been studied and then void nucleation and growth kinetics have been examd. A nucleation criterion of voids at the .alpha./.beta. interface, based on plastic strain and hydrostatic stress, has been identified for each whereas voids are **created** in the .alpha.-phase for a const. plastic strain. Void no. and length increase exponentially with resp. strain and triaxiality. They both drastically increase for crit. couples of strain and triaxiality. Three types of fracture occur depending on strain and triaxiality. The first, assisted by triaxiality, leads to fracture by void growth. The second appears for less triaxiality and under a certain amt. of strain. In this case, the quantity of voids is the cause of failure. At last, when the stress triaxiality is too low, no voids are **created** during loading but plastic strain localizes in shearing **bands** and the .alpha.-grains rotate so as to accommodate more plastic strain. The materials then develop a plastic strain instability at a macroscopic scale. These types of fracture were reported on fracture maps for the different alloys studied. The internal stress (X) is directly linked to the triaxiality level needed for void nucleation. So, this mech. parameter greatly influences the fracture modes experienced by the different materials. Besides, calcd. loading paths of points located ahead of the crack tip can be plotted on the fracture maps. Only a point located at a certain distance from the crack tip experiences enough strain and triaxiality to intersect the crit. growth curve. This distance from the crack corresponds to the exptl. zone where damage has been obsd. on an interrupted CT sample for a charge slightly before fracture. **Toughness** of the titanium alloys studied can be predicted as far as the fracture maps and loading paths ahead of the CT crack are detd.

CC 56-12 (Nonferrous Metals and Alloys)

IT 12743-70-3, TA6V **52293-96-6**, Ti6-2-4-6 185402-05-5, TD5AC

RL: PEP (Physical, engineering or chemical process); PRP (Properties);  
PROC (Process)

(fracture maps of titanium alloys)

IT **52293-96-6**, Ti6-2-4-6

RL: PEP (Physical, engineering or chemical process); PRP (Properties);  
PROC (Process)

(fracture maps of titanium alloys)

L63 ANSWER 6 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1997:65744 Document No. 126:92972 The effect of hot working on the transformation microstructure of the titanium alloy Ti-17. Rowe, R. G.; Sundell, R. E.; Gigliotti, M. F. X. (GE Corporate Research and Development, Schenectady, NY, 12309, USA). Titanium '95: Science and Technology, Proceedings of the World Conference on Titanium, 8th, Birmingham, UK, Oct. 22-26, 1995, Meeting Date 1995, Volume 3, 2250-2257. Editor(s): Blenkinsop, P. A.; Evans, W. J.; Flower, H. M. Institute of Materials: London, UK. (English) 1996. CODEN: 63XBAB.

AB The microstructure of beta forged, soln. heat treated, and aged Ti-17 consists of coarse Widmanstaetten alpha plates in a matrix of "aged beta" within prior beta grains. This microstructure has a good balance of properties such as **creep** and **fatigue fracture resistance**. However, Ti-17, thermomechanically processed to thin sections, cools rapidly, and a fine .alpha.+beta. **lath structure** with low fracture **toughness** is formed. The control of cooling rate after forging offers a means of **producing** coarse **lath** microstructures without re-soln. heat treatment. Investigation of cooling rate effects allowed an opportunity to det. if prior deformation affects the CCT diagram of Ti-17. It was found that there is an affect of forging on the high temp. transformation regime but that the effect on microstructure is diminished at lower temps.



- CC 56-8 (Nonferrous Metals and Alloys)  
IT 37329-07-0, Ti-17  
RL: PEP (Physical, engineering or chemical process); PRP (Properties);  
PROC (Process)  
(effect of hot working on transformation microstructure of Ti alloy Ti-17)
- IT 37329-07-0, Ti-17  
RL: PEP (Physical, engineering or chemical process); PRP (Properties);  
PROC (Process)  
(effect of hot working on transformation microstructure of Ti alloy Ti-17)
- L63 ANSWER 7 OF 15 HCAPLUS COPYRIGHT 2002 ACS  
1996:74892 Document No. 124:124120 Creep behavior of cast TiAl based intermetallics. Kim, S.; Cho, W.; Hong, C.-P. (Dep. Metallurgical Engineering, Yonsei Univ., Seoul, S. Korea). Mater. Sci. Technol., 11(11), 1147-55 (English) 1995. CODEN: MSCTEP. ISSN: 0267-0836.
- AB Const. load tensile creep tests were carried out on the cast TiAl based intermetallics Ti-47Al-2Mn, Ti-47Al-2Zr, and Ti-48Al (at.-%), **prepd.** by plasma arc melting. Two microstructural conditions dependent on heat treatment were evaluated as follows: a fully lamellar (FL) scheme consisting of a fully transformed coarse lamellar **structure** with **.alpha.2 lath** plus **.gamma.** **lath** within the grain interiors; and a duplex scheme consisting of fine equiaxed grains of **.gamma.** with **.alpha.2/.gamma.** lamellae. The steady state creep behavior of both microstructural conditions, for each compn., was studied under stresses of 70-300 MNm<sup>-2</sup> in the temp. range 700-900.degree.C. The microstructure was found to have a pronounced influence on the **creep resistance**. The FL microstructure exhibited superior **creep resistance** to the duplex microstructure. At temps. and stress levels at which direct comparisons can be **made**, the steady state creep rates of the FL structures are an order of magnitude lower than those of the duplex structure. The apparent creep activation energies and stress exponents were measured for both microstructural conditions for each compn. The temp. and stress dependence of the steady state creep rate of both microstructures can be described by the power law creep equation, suggesting dislocation motion as the operative deformation mechanism.
- CC 56-12 (Nonferrous Metals and Alloys)  
IT 159222-73-8, Aluminum 47, titanium 51, zirconium 2 (atomic)  
RL: PRP (Properties)  
(intermetallic; creep behavior of cast TiAl based intermetallics)
- IT 159222-73-8, Aluminum 47, titanium 51, zirconium 2 (atomic)  
RL: PRP (Properties)  
(intermetallic; creep behavior of cast TiAl based intermetallics)
- L63 ANSWER 8 OF 15 HCAPLUS COPYRIGHT 2002 ACS  
1991:518431 Document No. 115:118431 **Manufacture** of stainless steel **strip** having high strength and **toughness**. Takemoto, Toshihiko; Tanaka, Teruo; Murata, Yasushi (Nisshin Steel Co., Ltd., Japan). Jpn. Kokai Tokkyo Koho JP 02225647 A2 19900907 Heisei, 7 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1989-43237 19890227.
- AB The stainless steel for cold-rolled **strip** contains C .ltoreq.0.15, Si 3.0-7.0, Mn .ltoreq.8.0, Ni 8.0-13.0, Cr 12.0-17.0, and N .ltoreq.0.10% with the Ni equiv. of preferably 8-14%. The cold-rolled **strip** has martensitic **.alpha.'**-**phase structure**, and is heat treated at 600-900.degree. to give the **.gamma.-.alpha.'** final **microstructure**. Thus, the hot-rolled **strip** (contg. C 0.036, Si 3.02, Mn 1.05, Ni 8.92, Cr 15.33, and N 0.021%) was heated for solid soln., cold rolled at 60%

draft/pass, and then reheated at 710.degree. for 2 min. The resulting **strip** showed tensile strength of 115 kg/mm<sup>2</sup>, vs. 83 kg/mm<sup>2</sup> for the **strip** from SUS 304 stainless steel.

IC ICM C22C038-00

ICS C21D009-46; C22C038-40

CC 55-11 (Ferrous Metals and Alloys)

ST stainless steel **strip** heat treatment; silicon stainless steel **strip**

IT 135951-85-8 135951-86-9 135951-87-0 135951-88-1 135951-89-2  
135951-90-5 135951-91-6 135951-92-7 135951-93-8 135953-12-7  
135953-13-8 135953-14-9

RL: USES (Uses)

(high-strength **strip** from, by cold rolling and heat treatment)

IT 135953-12-7

RL: USES (Uses)

(high-strength **strip** from, by cold rolling and heat treatment)

L63 ANSWER 9 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1990:61152 Document No. 112:61152 Effect of structure on the physicommechanical properties of zirconium + 2.5% niobium alloy sheets. Bryukhanov, A. A.; Tarasov, A. F.; Goncharov, A. B.; Nerodenko, M. M. (Odessa, USSR). Izv. Akad. Nauk SSSR, Met. (6), 161-4 (Russian) 1989. CODEN: IZNMAQ. ISSN: 0568-5303.

AB The mech. properties after **strip** rolling, heat treatment, and recrystn. were detd. for **Zr-2.5% Nb alloy**, and were related to the texture of **strip** specimens 2 mm thick. Anisotropy of the mech. properties was present after the **strip** **manuf.**, and was retained after annealing in the **.alpha.-phase** range, but was removed by recrystn. annealing at 1000-1273 K in the **.beta.-phase** range. Impact **toughness** was decreased by cooling or quenching of recrystn. annealed specimens thus promoting the **formation** of an acicular martensitic phase.

CC 56-12 (Nonferrous Metals and Alloys)

ST zirconium niobium **strip** texture strength; recrystn zirconium niobium **toughness**

IT 50813-12-2, Niobium 2.5, zirconium 98

RL: PRP (Properties)

(mech. properties of, **strip** texture effect on, annealing in relation to)

IT 50813-12-2, Niobium 2.5, zirconium 98

RL: PRP (Properties)

(mech. properties of, **strip** texture effect on, annealing in relation to)

L63 ANSWER 10 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1988:154707 Document No. 108:154707 Laser beam beta heat treatment of **Zircaloy**. Sabol, G. P.; McDonald, S. G.; Nurminen, J. I.; Jacobsen, W. A. (Westinghouse R and D Cent., Pittsburgh, PA, 15235, USA). ASTM Spec. Tech. Publ., 939(Zirconium Nucl. Ind.), 168-86 (English) 1987. CODEN: ASTTA8. ISSN: 0066-0558.

AB A com. viable process of **.beta.-phase** heat treatment of **Zircaloy** tubing is developed, that circumvents the problems of dimensional stability at the treatment temp., protection from oxidn., and control of **.beta.** grain size. The method utilizes a high-power, geometrically shaped laser beam that impinges on the surface of the tubing as the tubing is rotated and translated through the focused beam. Beam energy, surface temp., and translation speed are chosen such that the zone of the **.beta.-phase** formation penetrates **.apprx.30%** of the wall thickness, and

the self-quenching provided by the underlying material produces a uniformly **rapid cooling** rate. The laser treatment is performed in Ar-filled chamber on next-to-final-sized tubing, and tube properties are recovered by **cold working** to final size and final **annealing**. Tubing of **Zircaloy-2** and **Zircaloy-4** produced by the laser beam .beta.-phase treatment is immune to nodules on both the inside and outside surfaces in the 24 h, 773 K steam test, and post-transition corrosion rates of **Zircaloy-4** are lower than those of conventionally processed tubing by factors of 2-3 at 633-673 K. The laser .beta. heat treatment process and the resultant tubing properties are described.

CC 56-5 (Nonferrous Metals and Alloys)

Section cross-reference(s): 73

IT **Nuclear reactor fuels and fuel elements**

(claddings, **Zircaloy**, laser .beta.-phase heat treatment of, corrosion resistance by)

IT 11068-94-3, **Zircaloy 2** 11068-95-4, **Zircaloy 4**

RL: USES (Uses)

(laser-beam .beta.-phase heat treatment of, corrosion resistance by)

IT 11068-94-3, **Zircaloy 2** 11068-95-4, **Zircaloy 4**

RL: USES (Uses)

(laser-beam .beta.-phase heat treatment of, corrosion resistance by)

L63 ANSWER 11 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1985:619509 Document No. 103:219509 A transition stress in the creep of an **alpha-phase** zirconium alloy at high temperature.

Donaldson, A. T.; Ecob, R. C. (Berkeley Nucl. Lab., CEGB, Berkeley/Gloucestershire, UK). Scr. Metall., 19(11), 1313-18 (English) 1985. CODEN: SCRMBU. ISSN: 0036-9748.

AB The strain-time curve **form** of PWR **Zircaloy-4**

[11068-95-4] fuel cladding creep at 923-1073 K had a transition at a temp.-dependent stress, .sigma.T, under const. loads. The transition stress value increases with decreasing temp. and is the macroscopic yield stress. **Creep** occurs by a **recovery**-controlled dislocation mechanism above .sigma.T, but at a lower stress a grain-boundary diffusion mechanism dominates deformation.

CC 56-12 (Nonferrous Metals and Alloys)

Section cross-reference(s): 71

ST **Zircaloy** cladding creep transition stress

IT **Nuclear reactor fuels and fuel elements**

(claddings, **Zircaloy**, creep transition stress at high temp.)

IT 11068-95-4

RL: USES (Uses)

(**nuclear reactor fuel cladding**, creep transition stress at high temp. in)

IT 11068-95-4

RL: USES (Uses)

(**nuclear reactor fuel cladding**, creep transition stress at high temp. in)

L63 ANSWER 12 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1982:604566 Document No. 97:204566 **Beta**-quenching of **Zircaloy** cladding tubes in intermediate or final size - methods to improve corrosion and mechanical properties. Andersson, T.; Vesterlund, G. (Sandvik AB, Sandviken, Swed.). ASTM Spec. Tech. Publ., 754(Zirconium Nucl. Ind.), 75-95 (English) 1982. CODEN: ASTTA8. ISSN: 0066-0558.

AB Three batches of **Zircaloy-2** [11068-94-3] tubing were

**.beta.**-quenched prior to the final cold-rolling, cold rolled 80 %, and **annealed** at 475 - 575.degree.. A 4th batch was **.beta.**-quenched in the final size. For comparison, std. tubing was included in all tests performed. The 2nd-phase particles were studied by means of optical and SEM. Corrosion testing was carried out at 400.degree. and in high-temp. (475 - 500.degree.) high-pressure steam. The mech. tests comprised tension, burst, and creep testing under internal pressure. **.beta.**-Quenching instead of an intermediate or the final **anneal** results in significant structural changes. The most striking features are the **formation** of a structure consisting of plates of **.alpha.-phase** and the repptn. of much finer 2nd-phase particles in the plate boundaries. Cold-rolling of **.beta.**-quenched hollows followed by a final **anneal** in the **.alpha.**-range will give an equiaxed structure, but the size and distribution of the 2nd phase obtained in **.beta.**-quenching will not be markedly changed. The wt. gain at 400.degree. increases slightly as a result of **.beta.**-quenching in intermediate or final size. In high-pressure steam at 475 - 500.degree., on the other hand, such **.beta.**-quenching has a dramatic beneficial effect on the corrosion resistance. The short-term strength as measured in tension and burst testing is improved by **.beta.**-quenching of hollows or finished tubes, whereas such treatment results in a slight drop in ductility, esp. for tubing **.beta.**-quenched in the final size. The 400.degree. transverse **creep strength** is increased by the introduction of **.beta.**-quenching prior to the final cold-rolling. The improvement is caused mainly by small 2nd-phase particles, **formed** during **.beta.**-quenching, which gives rise to pptn. hardening.

CC 71-5 (Nuclear Technology)

ST **Zircaloy** cladding **beta** quenching; reactor fuel cladding corrosion prevention

IT **Nuclear reactor fuels and fuel elements**  
(claddings, **Zircaloy**, **.beta.**-quenching of, for improved corrosion prevention and mech. properties)

IT **11068-94-3**  
RL: PROC (Process)  
(**.beta.**-quenching of cladding tubes of, for improved corrosion prevention and mech. properties)

IT **11068-94-3**  
RL: PROC (Process)  
(**.beta.**-quenching of cladding tubes of, for improved corrosion prevention and mech. properties)

L63 ANSWER 13 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1978:178939 Document No. 88:178939 Nodular corrosion of the **Zircaloys**. Johnson, A. B., Jr.; Horton, R. M. (Corros. Res. Eng. Sect., Battelle Northwest, Richland, Wash., USA). ASTM Spec. Tech. Publ. (STP 633, Zirconium Nucl. Ind.), 295-311 (English) 1977. CODEN: ASTTA8. ISSN: 0066-0558.

AB Oxide nodules form on **Zircaloy** nuclear components under irradiation. Similar nodules were obsd. on **Zircaloy** coupons in cold-rolled or extruded conditions after autoclave treatments at 475 and 500.degree. in steam at 1500-1700 psi. The stages of nodular corrosion in the autoclave were: nodule nucleation, growth, coalescence, propagation to accelerated uniform corrosion, and complete specimen oxidn. Observations on boiling water reactor (BWR) fuel rods suggest that a similar progressive attack has occurred; however, in no case has the in-reactor attack appeared to progress to the stage of complete component failure. Recent autoclave tests confirmed the nodular character of the attack on **cold-worked** materials. Alpha **anneals** (up to 790.degree.) did

not suppress the nodular attack consistently. However, alpha + beta (840.degree.) and beta (1010 and 1040.degree.) **anneals** did suppress the attack if they were followed by a **fast cool**. The efficacy of the **anneals** applied similarly to **Zircaloy-2** [11068-94-3] and **Zircaloy-4** [11068-95-4]. Stresses assocd. with U-bend specimens and heavy (86 %) **cold work** did not enhance the nodular attack before stress relief occurred. The nodular attack on reactor components appears to depend on nuclear flux, and develops in oxygenated reactor coolants, principally in the vicinity of fuel rod spacers. Experience with irradiated specimens in reactor loops suggests that uniform concns. of dissolved O alone do not cause the large nodules which frequently develop on BWR fuel rods. Localized water chem. assocd. with flow disturbances or, in some cases, dissimilar metals in fuel spacers, may be factors in the nodular attack in-reactor.

CC 71-6 (Nuclear Technology)

Section cross-reference(s): 56

ST **Zircaloy** nodular corrosion reactor cladding; fuel cladding  
nodular corrosion

IT **Nuclear reactor fuels** and **fuel elements**  
(**claddings**, **Zircaloy**, nodular corrosion of)

IT 11068-94-3 11068-95-4

RL: PROC (Process)

(corrosion of reactor fuel claddings of, nodular)

IT 11068-94-3 11068-95-4

RL: PROC (Process)

(corrosion of reactor fuel claddings of, nodular)

L63 ANSWER 14 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1977:445639 Document No. 87:45639 The fatigue behavior of **.alpha** **-zirconium** and **Zircaloy-2** in the temperature range 20 to 700.degree.C. Snowden, K. U.; Stathers, P. A. (Mater. Div., Aust. At. Energy Comm. Res. Establ., Sutherland, Aust.). J. Nucl. Mater., 67(1-2), 215-28 (English) 1977. CODEN: JNUMAM.

AB Reverse plane-bending fatigue tests at 16 Hz were undertaken on Zr and **Zircaloy-2** at temps. between 20 and 700.degree. to study the influence of temp., environment, and cold work on fatigue life and the nature of fatigue damage. The fatigue limit (for 106 cycles) at 20 and 300.degree. was 138.6 .+- .24 and 107.4 .+- .16 MN/m2 resp. for crystal bar Zr, and 235.0 .+- .24 and 190.0 .+- .10 MN/m2 resp. for **Zircaloy-2**. In addn., push/pull fatigue tests on **Zircaloy-2** at 16 Hz showed that the fatigue limit (at 106 cycles) was 240 .+- .20 MN/m2 at 20.degree. and 175 .+- .20 MN/m2 at 300.degree. in good agreement with bending data. The temp. dependence of fatigue life at const. strain amplitude showed evidence of **fatigue strengthening** because of dynamic **strain** aging between 20 and 300.degree.. Above 300.degree., fatigue life decreased with increasing temp. Prior cold working by amts. up to 15% **strain** was detrimental to **fatigue resistance**. For **Zircaloy-2** at 20.degree., cold work reduced fatigue limit by .apprx.30%. Examn. of the microstructure of fatigue specimens revealed that the fatigue crack path was transcryst. at low temps. and intercryst. at high temps. The transition from a transcryst. to an intercryst. mode of failure occurred at .apprx.500.degree. for crystal bar Zr and .apprx.600.degree. for **Zircaloy-2**. The low-temp. mode of deformation was characterized by slip-band clustering, twinning, and slip-band extrusion. The high-temp. mode of deformation was characterized by fine slip, grain-boundary migration, and **formation** of a diamond grain structure.

CC 71-6 (Nuclear Technology)

Section cross-reference(s): 56, 75  
ST fatigue zirconium temp; **Zircaloy 2** fatigue temp; crystal transition zirconium fatigue temp; reactor fatigue zirconium; fuel fatigue zirconium  
IT **Nuclear** reactor fuels and fuel elements  
(fatigue behavior of zirconium and **Zircaloy 2** in relation to)  
IT 7440-67-7, properties **11068-94-3**  
RL: PRP (Properties)  
(fatigue of, temp. effect on)  
IT **11068-94-3**  
RL: PRP (Properties)  
(fatigue of, temp. effect on)

L63 ANSWER 15 OF 15 HCAPLUS COPYRIGHT 2002 ACS

1968:438267 Document No. 69:38267 Diffusion bonding in vanadium and zirconium. Vermani, S. K.; Murarka, S. P.; Agarwala, R. P. (Chem. Div., Bhabha At. Res. Centre, Bombay, India). India At. Energy Comm., Bhabha At. Res. Centre, BARC-303 6 pp. (English) 1967. CODEN: IABRAA.  
AB Investigations were carried out to study the possibility of using V as a protective coating over Zr since the H **formed** due to pyrolytic and radiolytic decompn. of org. coolants in nuclear reactors reacts with the cladding material, **Zr** and **Zr alloys**, and seriously affects the mech. properties. Even at low pressures, complete hydriding of Zr is possible. The clad can, however, be protected by coating it with a metal inert to H and having suitable nuclear and mech. properties. If such a coating be given, the problem of its compatibility with clad becomes important. Nuclear pure Zr, and V rods of 1/4-in. diam. were machined and abraded on carborundum and emery papers followed by electropolishing. Zr was electropolished in glacial HOAc and HClO<sub>4</sub> bath, and V in a 10% H<sub>2</sub>SO<sub>4</sub> bath. The polished ends of V and Zr samples were placed in contact with each other in a die locked by screwing the threaded cap. Some couples were **prepd.** by **annealing** to give diffusion bonding for nearly 10 hrs. at 400.degree. and for an hr. at 950.degree. in a purified He atm. The couples thus **formed** were further **annealed** in a controlled furnace at 400-1050.degree. in a purified He atm. at a pressure of 0.5 .times. 10<sup>-3</sup> mm. Hg. Air-quenched couples were sectioned perpendicular to the diffusion front and diffusion **bands** were measured with an accuracy of 10 .mu. on a microscope. At <800.degree., very little diffusion was observed in the .**alpha** phase of Zr. A photomicrograph of V-Zr couple **annealed** at 800.degree. for 34 days shows a narrow diffusion **band**, while the couple **annealed** at <800.degree. for longer periods did not show any diffusion **band** at all. In Zr at .gtoreq.900.degree., only one diffusion **band** was observed. It appears from the phase diagram of V-Zr system that 2 phases are possible. One is a compd. of definite compn. ZrV<sub>2</sub>, and other is .delta.-phase. The phase appearing in these couples is ZrV<sub>2</sub> and the absence of .delta.-phase can be attributed to the difficulty in distinguishing the phase under a microscope as this phase might have more or less a structure similar to that of V owing to high V content. The lack of appreciable diffusion in .**alpha**.-Zr is attributed to several reasons. There is no solid soly. of V in .**alpha**.-Zr while in the .**beta**.-phase, its increases with temp. to 1230.degree.. The low soly. of V in .**alpha**.-Zr is one of the controlling factors resulting in low diffusivity. The phase change of Zr from h.c.p. lattice in .**alpha**.-phase to b.c.c. lattice in the .**beta**.-phase increases the self-diffusion parameter. To what extent it affects is difficult to evaluate. No measurements were **made** to test the strength of the bond but the bonding appeared to be quite firm and can be explained by the fact that V and Zr **form** the terminal solid soln. The temp. of operation of a reactor is nearly

400.degree., at that temp. there will only be little, if any, interpenetration between V and Zr. No visible diffusion **band** appeared when a couple was **annealed** for 56 days at 400.degree.. At least from the compatibility point of view, V is a suitable barrier to protect Zr from the H embrittlement in org.-cooled reactors. 13 references.

CC 56 (Nonferrous Metals and Alloys)

IT **Nuclear** reactor **fuels**, uses and miscellaneous  
(**claddings**, diffusion coating of, for prevention of hydrogen embrittlement)

IT Coating materials  
(vanadium, on zirconium **nuclear** reactor **fuel**  
**cladding** materials, embrittlement by hydrogen in relation to)

IT 1333-74-0, properties  
RL: PRP (Properties)  
(embrittlement by, of zirconium **cladding** materials for  
**nuclear** reactor **fuels**, coating with vanadium for  
prevention of)

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L64 ANSWER 1 OF 5 HCAPLUS COPYRIGHT 2002 ACS

1997:389629 Document No. 127:98419 Effects of microstructure on the fracture **toughness** of Ti3Al-based titanium aluminides. Wu, X.; Bowen, P. (Sch. Metall. Mater., Univ. Birmingham, Birmingham, B15 2TT, UK). Metallurgical and Materials Transactions A: Physical Metallurgy and Materials Science, 28A(6), 1357-1365 (English) 1997. CODEN: MMTAEB. ISSN: 1073-5623. Publisher: Minerals, Metals & Materials Society.

AB The influence of microstructure on the fracture **toughness** of Ti-23Al-9Nb-2Mo-1Zr-1.2Si and Ti-23Al-11Nb-0.9Si (at.%) Ti3Al-based alloys has been investigated. Basket-weave microstructures comprising different vol. fractions of **.alpha.2** and retained **.beta.** phases were **produced** by systematic heat treatments. Besides the vol. fraction of the retained **.beta.**-phase, the av. size of the **.beta.-laths** has also been used to characterize these microstructures. The **toughness** of both alloys was examd. at room temp. The brittle transgranular fracture modes were controlled by microstructure. However, the **toughness** is not detd. solely by the vol. fraction retained **.beta.**-phase, and a linear relationship has been obtained between the fracture **toughness** and av. size of the retained **.beta.-laths**. The **toughness** of the alloys at room temp. is controlled primarily by the width of retained **.beta.-laths** rather than by the retained **.beta.-vol.** fraction.

CC 56-12 (Nonferrous Metals and Alloys)

ST titanium aluminide microstructure fracture **toughness**

IT Fracture **toughness**  
(effects of microstructure on fracture **toughness** of  
Ti3Al-based titanium aluminides)

IT 186003-49-6, Ti-23Al-11Nb-0.9Si 191978-05-9

RL: PRP (Properties)  
(effects of microstructure on fracture **toughness** of  
Ti3Al-based titanium aluminides)

IT 191978-05-9

RL: PRP (Properties)  
(effects of microstructure on fracture **toughness** of  
Ti3Al-based titanium aluminides)

L64 ANSWER 2 OF 5 HCAPLUS COPYRIGHT 2002 ACS

1992:412651 Document No. 117:12651 Mechanical properties, of powder

metallurgy Ti-829 and Ti-25Al-10Nb-3V-1Mo **produced** by rapid omnidirectional compaction. Osborne, N. R.; Porter, W. J.; Eylon, D. (Res. Inst., Univ. Dayton, Dayton, OH, USA). SAMPE Q., 22(4), 21-8 (English) 1991. CODEN: SAMQA2. ISSN: 0036-0821.

- AB Two powder-metallurgy high-temp. Ti alloys were compared. An evaluation of room-temp. tensile properties and high-temp. **creep strength** of IMI-829 near-.**alpha.** and Ti-25Al-10Nb-3B-1Mo Ti3Al alloys was performed. Prealloyed powders **produced** by the plasma rotating electrode process were compacted by rapid omnidirectional consolidation. Properly heat treated IMI-829 compacts performed as well as comparable near-.**alpha.** wrought alloys with similar microstructures. The as-compacted Ti3Al alloy showed substantially higher ductility in comparison with that of the wrought alloy. While the alloy demonstrated strength levels of .apprx.1100 MPa after heat treatment, the room temp. ductility was severely decreased to <1% under all heat treatment conditions. The Larsen-Miller comparison of the **creep strength** of the Ti3Al alloy placed it within the scatter **band** of values for the wrought alloy.
- CC 56-12 (Nonferrous Metals and Alloys)
- IT 69235-99-0, IMI 829 128867-71-0  
RL: PRP (Properties)  
(mech. properties of powder-metallurgy)
- IT 69235-99-0, IMI 829  
RL: PRP (Properties)  
(mech. properties of powder-metallurgy)

L64 ANSWER 3 OF 5 HCAPLUS COPYRIGHT 2002 ACS

1991:564403 Document No. 115:164403 Effect of cooling rate on mechanical properties in **Zircaloy-4** alloy. Jeong, Yong Hwan; Choi, Chong Sool; Rheem, Karp Soon (Dep. Metall. Eng., Yonsei Univ., Seoul, 120-749, S. Korea). Taehan Kumsok Hakhoechi, 29(2), 104-11 (Korean) 1991. CODEN: TKHCDJ. ISSN: 0253-3847.

- AB The effect of cooling rate on the mech. properties of **Zircaloy-4** alloy was studied for the specimens which were heated in the region of .**beta.**-phase and then cooled in various cooling media, such as ice brine, water, oil, air, and furnace atm. The ice brine and water quenching of the specimens resulted in higher **strength** and greater **strain** hardening rate than the oil quenching, air, and furnace cooling. The increase in the **strength** and **strain** hardening rate is attributed to the increase in stress required to move glide dislocations due to twins and tangled dislocations introduced during the quenching process, i.e., martensitic transformation. The **strength** and **strain** hardening rate were increased gradually as the cooling rate increased from furnace cooling (0.05.degree./s) to oil quenching (110.degree./s). The 2 properties are mainly controlled by .**alpha.**-lath size. From the microstructure and hardness, the ice brine and water quenched specimens resulted in faster recrystn. than the oil quenched and air cooled specimens. The ice brine quenched specimen was recrystd. through homogeneous nucleation, while the recrystn. of water-quenched specimen followed the bulge mechanism.
- CC 56-12 (Nonferrous Metals and Alloys)
- IT Martensitic structure  
(**formation** of, in zirconium alloy, cooling-rate effect on)
- IT 11068-95-4, **Zircaloy-4**  
RL: PRP (Properties)  
(mech. properties of, cooling rate effect on)
- IT 11068-95-4, **Zircaloy-4**  
RL: PRP (Properties)  
(mech. properties of, cooling rate effect on)



L64 ANSWER 4 OF 5 HCAPLUS COPYRIGHT 2002 ACS

1988:191111 Document No. 108:191111 Effect of substructure **formed** in prior .beta. grain on crack initiation and propagation **toughness** of Ti-6Al-2Sn-4Zr-6Mo alloy. Niinomi, Mitsuo; Inagaki, Ikuhiro; Kobayashi, Toshiro (Toyohashi Univ. Technol., Toyohashi, 440, Japan). Tetsu to Hagane, 74(3), 543-50 (Japanese) 1988. CODEN: TEHAA2. ISSN: 0371-6279.

- AB The instrumented Charpy impact test, static, and dynamic fracture **toughness** tests were carried out on Ti-6Al-2Sn-4Zr-6Mo alloy in which the prior .beta. grain size was changed by heat treatments. The elongation, crack initiation, and propagation **toughness** increased with the slight decrease in strength in the specimens with the increased prior .beta. grain size and prolonged soln. treatment time in the .beta. region. The crack propagation **toughness** increased remarkably. The colony size, the width of grain boundary .alpha. ., and the width and spacing of Widmanstaetten .alpha. also increased, but the subcolony spacing decreased with the increase in the prior .beta. grain size. The increase in the crack initiation **toughness** was mainly caused by the increase in the Widmanstaetten .alpha. lath or lath spacing. The increase in the crack propagation **toughness** was caused by the deflection of the crack pass, which was brought by the decrease in the intersubcolony spacing. The intersubcolony spacing decreased with the increase in the no. of .alpha. nucleation sites during diffusion-controlled .alpha. .fwdarw. .beta. transformation; such nucleation sites increased with the increase in the prior .beta. grain size. In such a situation, .alpha. nucleated in the interior of the .beta. grain and it increased its no. by the introduction of the working strain.
- CC 56-12 (Nonferrous Metals and Alloys)
- ST titanium alloy cracking **toughness** Widmanstaetten; aluminum titanium cracking **toughness** structure; tin titanium cracking **toughness** structure; zirconium titanium cracking **toughness** structure; molybdenum titanium cracking **toughness** structure
- IT Widmanstaetten structure  
(in titanium-aluminum-tin-zirconium-molybdenum alloy, crack initiation and propagation **toughness** in relation to)
- IT 52293-96-6, Ti6Al2Sn4Zr6Mo  
RL: USES (Uses)  
(crack initiation and propagation **toughness** of, substructure **formed** in .beta.-grains effect on)
- IT 52293-96-6, Ti6Al2Sn4Zr6Mo  
RL: USES (Uses)  
(crack initiation and propagation **toughness** of, substructure **formed** in .beta.-grains effect on)

L64 ANSWER 5 OF 5 HCAPLUS COPYRIGHT 2002 ACS

1976:466888 Document No. 85:66888 State of the surface layers of **manufactured** objects after polishing with diamond **strips**. Bekrenev, A. N.; Aleksentsev, E. I. (Kuibyshev, USSR). Sint. Almazy (6), 20-3 (Russian) 1975. CODEN: SIALBI.

- AB The cutting zone temps. for OT4 [12633-22-6], VT9 [12633-32-8], and VT20 [12670-26-7] Ti alloys were 620-730, 540-690, and 400-580.degree. during polishing with corundum [1302-74-5], Si carbide, and diamond **strips**, resp. Corresponding **fatigue strengths** of polished VT9 alloy were 48-50, 51-53, and 53-55 kg/mm2. The strength was 42-45 kg/mm2 prior to polishing. The .beta.-.alpha. transformation occurred during polishing. The extent was the highest during polishing with diamond **strips** because of low surface work hardening. Tensile stress developed in Cr bronze

[12672-11-6] contg. 0.8 wt.% Cr during polishing with corundum **strips**, and compressive stress developed during polishing with diamond **strips**.

CC 56-7 (Nonferrous Metals and Alloys)

IT 12633-22-6 **12633-32-8 12670-26-7**

RL: PROC (Process)

(polishing of, with corundum and diamond and silicon carbide, surface state in relation to)

IT 409-21-2, uses and miscellaneous

RL: USES (Uses)

(polishing with, of titanium alloys, effect on **fatigue strength**)

IT **12633-32-8 12670-26-7**

RL: PROC (Process)

(polishing of, with corundum and diamond and silicon carbide, surface state in relation to)

=> d L67 1-42 ti

L67 ANSWER 1 OF 42 HCAPLUS COPYRIGHT 2002 ACS

TI Influence of a zirconia layer on the mechanical behavior of **Zircaloy-4** cladding and thimble **tubes**

L67 ANSWER 2 OF 42 HCAPLUS COPYRIGHT 2002 ACS

TI Tensile test of hydrided **zircaloy**

L67 ANSWER 3 OF 42 HCAPLUS COPYRIGHT 2002 ACS

TI Evaluation of mechanical properties of hydrided cladding by using modified ring tensile test

L67 ANSWER 4 OF 42 HCAPLUS COPYRIGHT 2002 ACS

002 ACS

Cladding **tube** with zirconium liner for nuclear fuel rod and its \*manufacture\*\*\*

L67 ANSWER 6 OF 42 HCAPLUS COPYRIGHT 2002 ACS

TI Influence of cladding microstructure on the low enthalpy failures in RIA simulation tests

L67 ANSWER 7 OF 42 HCAPLUS COPYRIGHT 2002 ACS

TI Application of blasting-ball treatment in lengthening services life of some nuclear facilities

L67 ANSWER 8 OF 42 HCAPLUS COPYRIGHT 2002 ACS

TI Recent ABB BWR SVEA fuel failure experience

L67 ANSWER 9 OF 42 HCAPLUS COPYRIGHT 2002 ACS

TI Recent ABB BWR failure experience

L67 ANSWER 10 OF 42 HCAPLUS COPYRIGHT 2002 ACS

TI Zirconium alloy E635 as a material for fuel rod cladding and other components of VVER and RBMK cores

L67 ANSWER 11 OF 42 HCAPLUS COPYRIGHT 2002 ACS

TI BWR fuel secondary degradation status

- L67 ANSWER 12 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Reactor fuel cladding **pipe** with zirconium-liner and its **manufacture**
- L67 ANSWER 13 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Optimization of PWR behavior of stress-relieved **Zircaloy-4** cladding **tubes** by improving the **manufacturing** and inspection process
- L67 ANSWER 14 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Development of new zirconium alloys for PWR fuel rod cladding
- L67 ANSWER 15 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Development of new ferritic steels as cladding material for metallic fuel fast breeder reactor
- L67 ANSWER 16 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI **Manufacture** of zirconium-alloy **pipes** for **cladding nuclear fuels**
- L67 ANSWER 17 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI **Stress**-corrosion cracking **resistant** zirconium alloy **tubes** for **cladding of nuclear fuel**
- L67 ANSWER 18 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI **Stress**-corrosion cracking **resistant** zirconium alloy **tubes** for **cladding of nuclear fuel**
- L67 ANSWER 19 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI **Stress**-corrosion cracking **resistant** zirconium alloy **tubes** for **cladding of nuclear fuel**
- L67 ANSWER 20 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Fatigue behavior of neutron irradiated **Zircaloy-2** fuel cladding **tubes**
- L67 ANSWER 21 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI PWC-11 fuel pin development for SP-100
- L67 ANSWER 22 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Determination of plastic anisotropy of zirconium alloy cladding
- L67 ANSWER 23 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI **Manufacture** of zirconium alloy **tubes**
- L67 ANSWER 24 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Behavior of braze heat-affected zone in CANDU fuel sheaths
- L67 ANSWER 25 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Effect of chemical composition on corrosion resistance of **Zircaloy** fuel cladding **tube** for BWR
- L67 ANSWER 26 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Mechanical properties of zirconium alloy cladding **tubes** and critical fuel element power ramps
- L67 ANSWER 27 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Heat-resistant steels

- L67 ANSWER 28 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Nuclear fuel element
- L67 ANSWER 29 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI **Zircaloy tube** for **nuclear fuel cladding**
- L67 ANSWER 30 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Cladding **tube** for reactor fuel element
- L67 ANSWER 31 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Evaluation of the resistance of irradiated zirconium-liner cladding to iodine-induced stress corrosion cracking
- L67 ANSWER 32 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Fuel rods for nuclear reactors
- L67 ANSWER 33 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Iodine-induced stress corrosion cracking of copper-barrier **Zircaloy-4 tubes**
- L67 ANSWER 34 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Beta-quenching of **Zircaloy** cladding **tubes** in intermediate or final size - methods to improve corrosion and mechanical properties
- L67 ANSWER 35 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Rupture criterion for **Zircaloy** cladding, when swelling in a reactor transient, calculated from free energy conditions
- L67 ANSWER 36 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Is air a suitable environment for simulation of **Zircaloy** /steam-high temperature-oxidation within engineering experiments?
- L67 ANSWER 37 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI The effects of texture and surface condition on the iodine stress corrosion cracking susceptibility of unirradiated **Zircaloy-2**
- L67 ANSWER 38 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Behavior of unirradiated zirconium-lined and copper-plated **Zircaloy-2 tubing** under simulated PCI conditions
- L67 ANSWER 39 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Design of an irradiation device for the determination of the in-pile creep behavior of **Zircaloy** cladding **tubes** under internal and external overpressure, in FRG-2
- L67 ANSWER 40 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI Creep anisotropy of **Zircaloy** cladding **tubes**
- L67 ANSWER 41 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI **Fabrication** technology and quality for **Zircaloy** fuel-cladding **tubes**
- L67 ANSWER 42 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
TI **Fabrication** techniques and quality of **Zircaloy** cladding **tubes**

=> d L67 1-6,10,12-14,16-20, 23, 26,28-30, 32-34, 38-42 cbib abs hitind hitrn

L67 ANSWER 1 OF 42 HCAPLUS COPYRIGHT 2002 ACS

2002:392512 Document No. 137:12162 Influence of a zirconia layer on the mechanical behavior of **Zircaloy-4** cladding and thimble **tubes**. Berat-Robert, Laurence; Pelchat, Jacques; Limon, Roger; Maury, Roger; Pele, Jacques; Cappelaere, Chantal; Prioul, Claude; Bouffioux, Pol; Diz, Jesus (Commissariat a l'Energie Atomique/ Saclay, Gif sur Yvette Cedex, 91191, Fr.). Proceedings of the International Topical Meeting on LWR Fuel Performance, 10th, Park City, UT, United States, Apr. 10-13, 2000, 90-100. American Nuclear Society: La Grange Park, Ill. ISBN: 0-89448-656-X (English) 2000. CODEN: 69CPAO.

AB For std. PWR fuel assemblies, in reactor conditions, the microstructure of **Zircaloy-4** alloy is progressively modified by irradiation. Cladding and thimble **tubes** also undergo corrosion: oxidation and hydriding. This phenomenon induces a reduction of the safe metal thickness and may involve changes of the mechanical properties of the cladding. This work aims at improving the knowledge and understanding of the zirconia effect in the mechanical behavior of **Zircaloy-4** claddings under reactor and dry storage conditions. Up to now, irradiation, hydriding and oxidizing effects have not often been studied separately. That is why the study only focuses on the oxidizing effects. To simplify testing conditions, unirradiated samples taken from recrystallized **Zircaloy-4** thimble and cladding **tubes** are used. They are previously oxidized under CO<sub>2</sub> atmosphere. In order to determine the effect of the oxide behavior of the **Zircaloy-4 tubes** for different mechanical loadings, several kinds of mechanical tests are carried out: axial tensile tests, axial creep tests, burst tests and internal pressure creep tests. For these tests, samples with various oxide thicknesses are used (between 1 and 30 µm on each side). Major reinforcement due to the oxide layer occurs under axial loading. For example, for axial creep tests, the measured strain after 10 days for a **tube** with a 10 µm oxide thickness is 4 times less than for the unoxidized specimen. The same reinforcement phenomenon is observed at the beginning of axial tensile tests. Tests under bi-axial loading do not reveal so much influence by the oxide layer. The circumferential plastic strain of an oxidized **tube** is higher than the strain of an unoxidized one. During burst tests, for example, an oxidized **tube** provides results comparable to those obtained with a **tube** whose thickness corresponds to the remaining unoxidized cladding thickness. After some mechanical tests, a local characterization of the oxide layer is achieved by S.E.M. observations to elucidate zirconia behavior and to establish which mechanisms influence oxidized **tube** deformation. For samples tested under axial creep load, the oxide layers are cracked perpendicular to the applied load. The zirconia layer reduces the **tube** creep and the **Zircaloy** strain seems to be closely controlled by oxide cracking. In the case of samples tested under internal pressure creep load, the oxide layers are largely cracked parallel to the **tube** axis, that is perpendicular to the maximal principal stress. The oxidized **tubes** behave like thinner **tubes** because of the extensive embrittlement of the oxide layer. These different tests underline the fact that the influence of the oxide layer varies according to the type of applied load. On the one hand, in the case of axial loading, zirconia has a beneficial influence on cladding mechanical resistance and reduces creep mechanism. However, zirconia does not demonstrate such a favorable effect under bi-axial loading.

CC 71-5 (Nuclear Technology)

Section cross-reference(s): 56

ST zirconia **Zircaloy** cladding thimble **tube** PWR fuel assembly

- IT Fuel assemblies  
(PWR; influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes** in)
- IT Plastic deformation  
(circumferential plastic strain of oxidized **tube** is higher than strain of unoxidized one.)
- IT Pressurized water nuclear reactors  
(fuel assemblies; influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes** in)
- IT Mechanical properties  
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**)
- IT **Nuclear fuel element cladding**  
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**.)
- IT Microstructure  
(microstructure of **Zircaloy-4** alloy is progressively modified by irradiation.)
- IT Embrittlement  
(oxidized **tubes** behave like thinner **tubes** because of extensive embrittlement of oxide layer.)
- IT 1314-23-4, Zirconia, **formation** (nonpreparative)  
RL: FMU (Formation, unclassified); FORM (Formation, nonpreparative)  
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**)
- IT **11068-95-4, Zircaloy-4**  
RL: PRP (Properties)  
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**)
- IT **11068-95-4, Zircaloy-4**  
RL: PRP (Properties)  
(influence of zirconia layer on mech. behavior of **Zircaloy-4** cladding and thimble **tubes**)

L67 ANSWER 2 OF 42 HCAPLUS COPYRIGHT 2002 ACS

2002:94997 Document No. 136:174308 Tensile test of hydrided **zircaloy**  
. Kuroda, Masatoshi; Yamanaka, Shinsuke; Setoyama, Daigo; Uno, Masayoshi; Takeda, Kiyoko; Anada, Hiroyuki; Nagase, Fumihisa; Uetsuka, Hiroshi  
(Department of Nuclear Engineering, Graduate School of Engineering, Osaka University, Suita, 565-0871, Japan). Journal of Alloys and Compounds, 330-332, 404-407 (English) 2002. CODEN: JALCEU. ISSN: 0925-8388.  
Publisher: Elsevier Science S.A..

AB To examine the influence of pptd. Zr hydride on the failure behavior and **fracture strength** of light H2O reactor (LWR) cladding **tubes**, tensile tests were performed at room temp. for nonhydrided and hydrided **Zircaloy** sheet-type specimens with gauge section of 10.0.times.5.0 mm and thicknesses of 0.5, 1.0, 1.5, 2.0, 2.5, and 3.0 mm. For specimens with thickness >2.5 mm, the ultimate tensile strength of the specimens appeared to be independent of thickness, which implied that plane strain condition was attained. For the specimen with 2.5 mm thickness, the ultimate tensile strength increased slightly with increasing av. H concn. Through microscopic observation of the hydrided specimen surface by SEM, matrix/hydride de-bonding was not **generated** but micro-cracks perpendicular to the axial direction were **produced** at the hydride layer.

- CC 71-3 (Nuclear Technology)  
Section cross-reference(s): 56
- ST tensile hydrided **zircaloy** LWR cladding
- IT Light-water nuclear reactors  
**Nuclear fuel element cladding**

- (failure behavior and **fracture strength** of light H2O reactor cladding)
- IT Fracture (materials)  
(influence of pptd. Zr hydride on failure behavior and **fracture strength**)
- IT Tensile strength  
(tensile test of hydrided **zircaloy**)
- IT **11068-95-4**  
RL: PRP (Properties)  
(Tensile test of hydrided **zircaloy**)
- IT 11105-16-1, Zirconium hydride  
RL: PRP (Properties)  
(influence of pptd. Zr hydride on failure behavior and **fracture strength**)
- IT 12184-88-2, Hydride  
RL: PEP (Physical, engineering or chemical process); PROC (Process)  
(tensile test of hydrided **zircaloy**)
- IT **11068-95-4**  
RL: PRP (Properties)  
(Tensile test of hydrided **zircaloy**)
- L67 ANSWER 3 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
2002:20069 Document No. 136:223054 Evaluation of mechanical properties of hydrided cladding by using modified ring tensile test. Kitano, Koji; Fuketa, Toyoshi; Uetsuka, Hiroshi (Dep. Reactor Safety Res., Nuclear Safety Res. Center, Tokai Res. Establishment, Japan Atomic Energy Research Institute, Tokai-mura, Naka-gun, Ibaraki-ken, Japan). JAERI-Research, 2001-041, i-iv, 1-24 (Japanese) 2001. CODEN: JERIE4.
- AB Results from the pulse irradiation tests at NSRR indicated that failure of high burn-up fuel rod under Reactivity Initiated Accident conditions occurs due to pellet/cladding mech. interaction (PCMI). The authors performed modified ring tensile test on cladding **tube** samples with artificially **made** hydride rim to evaluate the influence of hydride rim on mech. properties of cladding in hoop direction. Fracture strain reduces with hydride rim thickness at room temp. because cracks could **generate** in brittle hydride rim region at the beginning stage of deformation. At elevated temp. (573 K), fracture strain varied depending not only on thickness of hydride rim but also on hydride d. in rim region. The specimen with hydride rim of low hydride d. showed larger fracture strain regardless of the hydride rim thickness. This may be attributed to ductile-brittle transition of hydride rim region with temp. increase. The rim of low hydride d. could be not brittle but ductile at 573 K. Thus probably fracture strain of the specimen with thick hydride rim becomes larger when the hydride d. in rim region is low.
- CC 71-5 (Nuclear Technology)  
Section cross-reference(s): 56
- ST hydride **nuclear** reactor **fuel cladding**  
tensile strain
- IT Fracture **toughness**  
**Nuclear fuel** element **cladding**  
Nuclear reactor accident  
Strain  
Tensile strength  
(evaluation of mech. properties of hydrided cladding by using modified ring tensile test)
- IT **11068-95-4, Zircaloy 4**  
RL: PRP (Properties)  
(evaluation of mech. properties of hydrided cladding by using modified ring tensile test)
- IT **11068-95-4, Zircaloy 4**

RL: PRP (Properties)

(evaluation of mech. properties of hydrided cladding by using modified ring tensile test)

L67 ANSWER 4 OF 42 HCAPLUS COPYRIGHT 2002 ACS

2000:808616 Document No. 133:341714 Cladding for use in nuclear reactors having improved resistance to cracking and corrosion. Admson, Ronald Bert; Lutz, Daniel Reese; Marlowe, Mickey Orville; Schardt, John Frederick; Williams, Cedric David (General Electric Company, USA). Eur. Pat. Appl. EP 1052650 A1 20001115, 16 pp. DESIGNATED STATES: R: AT, BE, CH, DE, DK, ES, FR, GB, GR, IT, LI, LU, NL, SE, MC, PT, IE, SI, LT, LV, FI, RO. (English). CODEN: EPXXDW. APPLICATION: EP 2000-304016 20000512. PRIORITY: US 1999-312021 19990514.

AB An improved fuel element for use in a nuclear reactor comprised of a central core of nuclear material, which is surrounded by a composite cladding. The cladding has an outer metallic **tubular** portion comprised of well-known cladding alloys used for such purposes. Metallurgically bonded to the outer metallic **tubular** portion is a com. pure zirconium microalloyed with a controlled quantity of iron. The zirconium microalloyed with iron **produce** an inner metallic barrier having a beneficial balance between **stress** corrosion crack **resistance** and corrosion resistance while retaining other beneficial properties of pure zirconium, such as ductility.

IC ICM G21C003-07

ICS G21C003-20

CC 71-5 (Nuclear Technology)

ST **cladding nuclear fuel** resistance cracking  
corrosion zirconium alloy iron

IT **Nuclear fuels**

Nuclear reactors

Stress corrosion cracking

(**cladding** for use in nuclear reactors having improved resistance to cracking and corrosion)

IT **11068-94-3, Zircaloy 2**

RL: NUU (Other use, unclassified); USES (Uses)  
(use in nuclear reactor fuel elements)

IT **11068-94-3, Zircaloy 2**

RL: NUU (Other use, unclassified); USES (Uses)  
(use in nuclear reactor fuel elements)

L67 ANSWER 5 OF 42 HCAPLUS COPYRIGHT 2002 ACS

2000:585568 Document No. 133:184582 Cladding **tube** with zirconium liner for nuclear fuel rod and its **manufacture**. Nakatsukasa, Masafumi (Japan Nuclear Fuel Development Co., Ltd., Japan; Toshiba Corp.; Hitachi, Ltd.). Jpn. Kokai Tokkyo Koho JP 2000230993 A2 20000822, 6 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1999-30795 19990209.

AB The cladding **tube** comprises a **Zr alloy**

**tube** with a high-purity Zr liner contg. 400-1000 ppm (in total) Fe and/or Cr and .ltoreq.400 ppm O as additives and .ltoreq.1000 ppm inevitable impurities. Av. crystal grain size of the Zr liner is 10-13 .mu.m, and at least part of the inside of the liner has oxidized layer.

In **manufg.** the cladding **tube** by inserting a Zr

**pipe** into a **Zr alloy tube**, bonding

by metallurgy, and alternately repeating cold rolling and annealing,

oxidn. of the cladding **tube** is carried out in finish annealing

process. The cladding **tube** has high **resistance** to

**stress** corrosion cracking, and is useful for light-water nuclear reactors (LWR).

IC ICM G21C003-20

ICS G21C003-06



- CC 71-4 (Nuclear Technology)  
Section cross-reference(s): 56
- ST zirconium liner **cladding tube nuclear fuel rod**; oxidn annealing zirconium **cladding tube nuclear fuel**; **stress corrosion cracking resistance cladding tube nuclear fuel**
- IT **Nuclear fuels**  
(**Zr alloy cladding tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress corrosion cracking** for nuclear fuel rod)
- IT Oxidation  
(annealing and oxidn. in **manuf.** of **Zr alloy cladding tube** with **Zr** liner for nuclear fuel rod)
- IT Coating materials  
(anticorrosive; **Zr alloy cladding tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress corrosion cracking** for nuclear fuel rod)
- IT Coating materials  
(linings; **Zr alloy cladding tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress corrosion cracking** for nuclear fuel rod)
- IT Containers  
(**tubes**; **Zr alloy cladding tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress corrosion cracking** for nuclear fuel rod)
- IT 7440-67-7, Zirconium, uses 11068-94-3 12614-57-2 141825-38-9  
RL: DEV (Device component use); PEP (Physical, engineering or chemical process); PRP (Properties); PROC (Process); USES (Uses)  
(**Zr alloy cladding tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress corrosion cracking** for nuclear fuel rod)
- IT 7782-44-7, Oxygen, uses  
RL: MOA (Modifier or additive use); USES (Uses)  
(microalloying element; **Zr alloy cladding tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress corrosion cracking** for nuclear fuel rod)
- IT 11068-94-3 12614-57-2 141825-38-9  
RL: DEV (Device component use); PEP (Physical, engineering or chemical process); PRP (Properties); PROC (Process); USES (Uses)  
(**Zr alloy cladding tube** with **Zr** liner contg. Fe and/or Cr and O for high **resistance** to **stress corrosion cracking** for nuclear fuel rod)
- L67 ANSWER 6 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
2000:242474 Document No. 133:10033 Influence of cladding microstructure on the low enthalpy failures in RIA simulation tests. Garde, A. M. (ABB Combustion Engineering Nuclear Fuel, Windsor, CT, 06095, USA). ASTM Special Technical Publication, STP 1354(Zirconium in the Nuclear Industry: Twelfth International Symposium, 1998), 234-255 (English) 2000. CODEN: ASTTA8. ISSN: 0066-0558. Publisher: ASTM.
- AB A review with 30 refs. Welding used during the **prepn.** of specimens for Reactivity Initiated Accident (RIA) simulation testing from fuel rods previously irradiated in reactors introduces the following 4 microstructural changes in the specimen: (a) annealing of irradiation damage, (b) a change from an .alpha.-phase structure of **Zircaloy** to a

transformed .beta. structure in the cladding, (c) dissoln. of the hydride rim **formed** under the oxide on the cladding **tube** outer surface during normal irradiation and possible radial-oriented hydride reprecipitation at the transformed .beta. platelet boundaries, and (d) reprecipitation of 2nd-phase particles previously dissolved due to radiation damage. A 5th factor, the change in the texture of **Zircaloy**, is also introduced due to the welding operation. The possible effect of these 5 changes on the specimen fracture **toughness** and failure enthalpy is evaluated. The published data on the mechanical properties of irradiated and unirradiated transformed beta structure of **Zircaloy** charged with H are reviewed to evaluate the impact of the 5 anticipated microstructural changes on the failure enthalpy. The available RIA simulation test fracture data with low-failure enthalpy are reviewed. The limited failure path information available appears to indicate that microstructural factors have contributed to the low-enthalpy failures. The applicability of the results from the low-enthalpy RIA test failures to Light Water Reactor (LWR) nuclear fuel should be based on the representativeness of the RIA specimen microstructure to that of the LWR fuel cladding.

CC 71-0 (Nuclear Technology)

Section cross-reference(s): 56

IT Fracture **toughness**

Light-water nuclear reactors

**Nuclear fuel element cladding**

Nuclear reactor accident

Simulation and Modeling, physicochemical

Structural phase transition

Welding of metals

(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

IT 11068-94-3, **Zircaloy-2** 11068-95-4,  
**Zircaloy-4**

RL: PRP (Properties)

(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

IT 11068-94-3, **Zircaloy-2** 11068-95-4,  
**Zircaloy-4**

RL: PRP (Properties)

(influence of cladding microstructure on low enthalpy failures in RIA simulation tests)

L67 ANSWER 10 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1997:71449 Document No. 126:230459 Zirconium alloy E635 as a material for fuel rod cladding and other components of VVER and RBMK cores. Nikulina, Antonina V.; Markelov, Vladimir A.; Peregud, Mikhail M.; Bibilashvili, Yuriy K.; Kotrekhov, Vladimir A.; Lositsky, Anatoly F.; Kuzmenko, Nikolay V.; Shevnin, Yuriy P.; Shamardin, Valentin K.; Kobylansky, Gennady P.; Novoselov, Andrey E. (A. A. Bochvar All-Russia Scientific Res. Inst. Inorganic Materials, Moscow, 123060, Russia). ASTM Special Technical Publication, STP 1295 (Zirconium in the Nuclear Industry: Eleventh International Symposium, 1995), 785-804 (English) 1996. CODEN: ASTTA8. ISSN: 0066-0558. Publisher: American Society for Testing and Materials.

AB Data are given on **Zr alloy E635 (Zr**

-1.2Sn-1Nb-0.4Fe), developed in Russia as a fuel rod cladding and other component material for use in cores of WWR and RBMK types. The alloy is much superior to binary alloys with 1.0 and 2.5% Nb and **Zircaloys** in terms of its **resistance** to irradiation-induced **creep** and growth and nodular corrosion. The creep rate of the alloy is slightly dependent on irradiation temperature, stress, neutron fluence, and neutron dose. The alloy is subject to substantial irradiation hardening while retaining its high-percent

elongation. Corrosion, **creep**, and growth **resistances** are slightly dependent on the structure of components (alloy, final **product**). Based on the previously studied influence of impurities, structure, heat treatment, and working schedules, the technol. processes were designed and mastered com. for **fabrication** of **tubes**, bars, strips, and fuel rod claddings from this alloy. Components are **produced** com. Fuel assemblies with fuel rods clad in the E635 alloy were successfully tested in the RBMK reactor at the Leningrad NPP as well as in exptl. reactors under WWR-1000 conditions. Today, the E635 alloy is recommended as a promising material for use in cores of WWR-1000 and WWR of new **generations** as well as RBMK-type reactors having a longer fuel cycle.

CC 71-5 (Nuclear Technology)

Section cross-reference(s): 56

IT **Nuclear fuel element cladding**

(zirconium alloy E635 as material for fuel rod cladding and other components of VVER and RBMK cores)

IT 150287-03-9, E635

RL: NUU (Other use, unclassified); PRP (Properties); USES (Uses)  
(zirconium alloy E635 as material for fuel rod cladding and other components of VVER and RBMK cores)

IT 150287-03-9, E635

RL: NUU (Other use, unclassified); PRP (Properties); USES (Uses)  
(zirconium alloy E635 as material for fuel rod cladding and other components of VVER and RBMK cores)

L67 ANSWER 12 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1995:982773 Document No. 124:39886 Reactor fuel cladding **pipe** with zirconium-liner and its **manufacture**. Nakatsuka, Masafumi (Nippon Kakunenryo Kaihatsu Kk, Japan). Jpn. Kokai Tokkyo Koho JP 07248391 A2 19950926 Heisei, 4 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1994-42388 19940314.

AB The cladding **pipe** has an inside structure consisting of a **Zr alloy** lining and a layer preventing H-permeation. The **pipe** may be coated with the layer preventing H-permeation. The layer may be black monoclinic Zr oxide. In the **manuf.**, the layer is obtained by oxidn. the **pipe** in a vapor at atm. pressure. The **pipe** shows good **resistance** to H-**embrittlement**.

IC ICM G21C003-20

ICS G21C003-06

CC 71-5 (Nuclear Technology)

Section cross-reference(s): 56

ST reactor fuel **pipe** zirconium lining; hydrogen

**embrittlement resistance** zirconium **pipe**

IT **Nuclear reactor fuels and fuel elements**

(**claddings**, reactor fuel cladding **pipe** with Zr-liner and Zr oxide and its **manuf.**)

IT 1314-23-4P, Zirconium oxide, uses

RL: PNU (Preparation, unclassified); TEM (Technical or engineered material use); PREP (Preparation); USES (Uses)  
(reactor fuel cladding **pipe** with Zr-liner and Zr oxide and its **manuf.**)

IT 7440-67-7, Zirconium, uses 11068-94-3, Zircaloy-2

RL: TEM (Technical or engineered material use); USES (Uses)  
(reactor fuel cladding **pipe** with Zr-liner and Zr oxide and its **manuf.**)

IT 11068-94-3, Zircaloy-2

RL: TEM (Technical or engineered material use); USES (Uses)  
(reactor fuel cladding **pipe** with Zr-liner and Zr oxide and

its manuf.)

- L67 ANSWER 13 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
1995:700115 Document No. 123:239819 Optimization of PWR behavior of stress-relieved **Zircaloy-4** cladding **tubes** by improving the **manufacturing** and inspection process. Mardon, Jean-Paul; Charquet, Daniel; Senevat, Jean (FRAMATOME Nuclear Fuel Division, FRAGEMA, Lyon, 69006, Fr.). ASTM Spec. Tech. Publ., STP 1245(Zirconium in the Nuclear Industry: Tenth International Symposium, 1993), 328-48 (English) 1994. CODEN: ASTTA8. ISSN: 0066-0558.
- AB With the aim of optimizing the basic properties of stress-relieved **Zircaloy-4** cladding **tubes**, particularly those that **make** it possible to push back the initial technol. limits that may be encountered, and of reducing the scatter of those properties and enhancing **tube** quality, the role of the main parameters involved in **manufg.** the ingot, Trex, and cladding **tube** was evaluated on an industrial scale. Large-sized **tube** lots were **produced** under controlled **manufg.** conditions, then characterized by out-of-pile test results (short-and long-term corrosion, stress corrosion cracking (SCC), creep, mech., and structural properties) on finished **tubes**. For the studied parameters (chem. compn., no. of melt, quench rate, accumulated annealing parameter, the .SIGMA.A factor, surface condition (outside and inside diams.), and finished **tube** quality), this role is indeed important but complex due to the highly interactive nature of the variables studied. Adjustment of the chem. compn. within ASTM limits enables generalized corrosion resistance to be enhanced and irradiation growth to be minimized. A significant decrease of the observed scatter in corrosion and mech. properties is obtained by optimization of the .SIGMA.A range, the quenching rate, and the final heat treatment. The optimum seems to be reached for a final treatment at the highest possible temp. compatible with the stress-relieved state, corresponding to an av. ppt. size and .SIGMA.A. Also, by adding anneals upstream in the process, a further increase in this .SIGMA.A no longer seems to have a significant effect on generalized corrosion. Finally, extensive efforts were employed in the pickling, surface **prepn.** (outside diam. polishing, flush-pickling), and examn. method (UT, EC) leading to a sizable improvement in SCC resistance and to a reduction in scatter for finished **tubes**. The result of these optimizations was implemented in the current AFA-2G, that shows that under irradiation a 30% corrosion gain is reached after 3 cycles, without degrading **creep strength** or growth. The intrinsic effect of Sn on generalized corrosion resistance under irradiation was also confirmed on this occasion.
- CC 71-5 (Nuclear Technology)  
Section cross-reference(s): 56
- ST optimization PWR behavior stress relieved **Zircaloy**; cladding **tube** improving **Zircaloy** optimization
- IT **Nuclear reactor fuels and fuel elements**  
(**claddings**, Optimization of PWR behavior of stress-relieved **Zircaloy-4** cladding **tubes** by improving the **manufg.** and inspection process)
- IT 7440-31-5, Tin, processes 11068-95-4, **Zircaloy-4** 12586-31-1, Neutron  
RL: PEP (Physical, engineering or chemical process); PROC (Process)  
(Optimization of PWR behavior of stress-relieved **Zircaloy-4** cladding **tubes** by improving the **manufg.** and inspection process)
- IT 11068-95-4, **Zircaloy-4**  
RL: PEP (Physical, engineering or chemical process); PROC (Process)  
(Optimization of PWR behavior of stress-relieved **Zircaloy-**

4 cladding tubes by improving the manufg.  
and inspection process)

L67 ANSWER 14 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1994:665983 Document No. 121:265983 Development of new zirconium alloys for PWR fuel rod cladding. Mardon, Jean Paul; Charquet, Daniel; Senevat, Jean (Framatome Nuclear Fuel, Lyon, 69456, Fr.). Proc. Int. Top. Meet. Light Water React. Fuel Perform., 643-9. Am. Nucl. Soc.: La Grange Park, Ill. (English) 1994. CODEN: 60DGA9.

AB Within the scope of research into PWR fuel rod cladding materials with higher performance than **Zircaloy-4**, four zirconium alloys were **produced** and transformed into solid-wall and coextruded **tubes** according to industrial process outlines. These alloys break down into two groups : three ultra low tin alloys contg. (Fe, Cr, Nb, O or V) and one ternary niobium - oxygen alloy. These solns. were characterized out of pile for corrosion, creep, SCC, mech. and structural properties, high-temp. creep, and in pile for corrosion, growth and creep. In relation to stress-relieved AFA-2G **Zircaloy-4** (low tin optimized **Zircaloy-4**), the new **Zr alloys** have exhibited better waterside corrosion resistance through three irradiation cycles (40 GWd/t). The results are not the same as those obtained in autoclave testing. The corrosion resistance found in autoclave testing is relatively smaller in the case of the ultra low tin alloys and higher for the ternary niobium - oxygen alloy. The monotonic influence from 0.5 % to 1.7 % tin on corrosion obsd. in autoclave is not seen under irradiation, it is more marked between 1.7 and 1.2 % than between 1.2 and 0.5 %. The corrosion behavior of the zirconium alloys outside **Zircaloy-4** cannot be detd. solely on the basis of autoclave tests in water or steam; final judgement has to be confirmed by irradiation with relatively high burnup data. The other properties important for clad behavior like creep and growth are at least identical to those of AFA-2G **Zircaloy-4** and significant improvements for these two properties are obsd. for some of these alloys. For the coextruded **tubes**, corrosion resistance is dictated by the external layer alloy; the substrate can provide SCC resistance and the substrate/external layer assembly provides mech. **strength** and **creep** performance.

CC 71-5 (Nuclear Technology)

ST zirconium alloy PWR fuel rod cladding; **nuclear reactor fuel cladding**

IT **Nuclear reactor fuels and fuel elements**  
(claddings, zirconium alloys for PWR)

IT **11068-95-4P, Zircaloy-4**

RL: PNU (Preparation, unclassified); PREP (Preparation)  
(PWR fuel rod cladding)

IT **81029-19-8P 158634-69-6P, Iron 0.2 niobium 0.5 oxygen**  
0.1 tin 0.5 zirconium 99 **158634-70-9P, Chromium 0.1 iron 0.2**  
oxygen 0.2 tin 0.5 zirconium 99 **158634-71-0P, Iron 0.6 oxygen**  
0.1 tin 0.5 vanadium 0.3 zirconium 99

RL: PNU (Preparation, unclassified); PRP (Properties); SPN (Synthetic preparation); PREP (Preparation)  
(PWR fuel rod cladding)

IT **11068-95-4P, Zircaloy-4**

RL: PNU (Preparation, unclassified); PREP (Preparation)  
(PWR fuel rod cladding)

IT **81029-19-8P 158634-69-6P, Iron 0.2 niobium 0.5 oxygen**  
0.1 tin 0.5 zirconium 99 **158634-70-9P, Chromium 0.1 iron 0.2**  
oxygen 0.2 tin 0.5 zirconium 99 **158634-71-0P, Iron 0.6 oxygen**  
0.1 tin 0.5 vanadium 0.3 zirconium 99

RL: PNU (Preparation, unclassified); PRP (Properties); SPN (Synthetic

preparation); PREP (Preparation)  
(PWR fuel rod cladding)

L67 ANSWER 16 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1993:134573 Document No. 118:134573 **Manufacture** of zirconium-alloy

**pipes for cladding nuclear fuels.**

Kikukawa, Tomokazu; Suda, Yoshitaka; Isobe, Takeshi (Mitsubishi Materials Corp., Japan). Jpn. Kokai Tokkyo Koho JP 04154943 A2 19920527 Heisei, 4 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1990-276862 19901016.

AB In the process, in which a **Zr-alloy pipe** is repeatedly extruded and annealed for recrystn., and is finally annealed for warp removal, the **pipe** is extruded while stress within the elastic limit of the alloy is applied in the axial direction of the **pipe**. The **Zr-alloy pipe** has increased **resistance to stress-corrosion cracking**.

IC ICM C22F001-18

ICS B21B021-00; G21C003-06

CC 71-5 (Nuclear Technology)

ST zirconium alloy **pipe cladding nuclear fuel**

IT **Nuclear reactor fuels and fuel elements**  
(**claddings, zirconium-alloy pipes, manuf. of**)

IT Zirconium alloy, base

RL: PROC (Process)

(**pipes from, for cladding nuclear fuels, manuf. of**)

IT 89342-04-1

RL: PROC (Process)

(**pipes from, for cladding nuclear fuels, manuf. of**)

IT 89342-04-1

RL: PROC (Process)

(**pipes from, for cladding nuclear fuels, manuf. of**)

L67 ANSWER 17 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1992:536009 Document No. 117:136009 **Stress-corrosion cracking**

**resistant zirconium alloy tubes for cladding**

**of nuclear fuel.** Mae, Yoshiharu; Isobe, Takeshi

(Mitsubishi Materials K. K., Japan). Jpn. Kokai Tokkyo Koho JP 04099256 A2 19920331 Heisei, 6 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1990-212642 19900810.

AB The title **tubes** are **manufd.** by extruding **Zr alloy** to give raw **tubes**, Pilger rolling and recrystn. annealing of the raw **tubes** for .gtoreq. 1 times, resp., then final Pilger rolling and stress-relief annealing for .gtoreq.1 times to give the o.d. redn. rate 1-15%. Thus, an extruded **tube** from **Zr alloy** contg. Sn 1.5, Fe 0.2, and Cr 0.1% after the heat treatment showed good machinability and high **stress-corrosion cracking resistance**.

IC ICM C22F001-18

ICS B21B021-00; B21C001-22; G21C003-06

CC 56-11 (Nonferrous Metals and Alloys)

Section cross-reference(s): 71

ST zirconium alloy **tube** cracking resistance; **stress corrosion cracking resistance tube; nuclear fuel cladding zirconium alloy tube**

IT **Pipes and Tubes**

(zirconium alloy, **stress-corrosion cracking-resistant**)

, manuf. of)

IT Nuclear reactor fuels and fuel elements  
(claddings, zirconium alloy tubes, good  
stress-corrosion cracking-resistant, manuf.  
of)

IT Zirconium alloy, base  
RL: PEP (Physical, engineering or chemical process); PROC (Process)  
(tubes, stress-corrosion cracking resistant  
, manuf. of, for cladding of nuclear  
fuel)

IT 89342-04-1P  
RL: PEP (Physical, engineering or chemical process); PREP (Preparation);  
PROC (Process)  
(tubes, stress-corrosion cracking resistant  
, manuf. of, for cladding of nuclear  
fuel)

IT 89342-04-1P  
RL: PEP (Physical, engineering or chemical process); PREP (Preparation);  
PROC (Process)  
(tubes, stress-corrosion cracking resistant  
, manuf. of, for cladding of nuclear  
fuel)

L67 ANSWER 18 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
1992:536008 Document No. 117:136008 Stress-corrosion cracking  
resistant zirconium alloy tubes for cladding  
of nuclear fuel. Mae, Yoshiharu; Isobe, Takeshi  
(Mitsubishi Materials K. K., Japan). Jpn. Kokai Tokkyo Koho JP 04099255  
A2 19920331 Heisei, 4 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP  
1990-212641 19900810.

AB The title tubes are manufd. by extruding of Zr  
alloy to give raw tubes, Pilger rolling of the raw  
tubes, and recrystn. annealing at 530-760.degree. under vacuum  
atm. for .gtoreq. 1 times, resp., then final Pilger rolling and  
stress-relief annealing at 430-490.degree. to give the o.d. redn. rate  
1-15 %. Thus, an extruded tube from Zr alloy  
contg. Sn 1.5, Fe 0.2, and Cr 0.1% after the heat treatment showed high  
stress-corrosion cracking resistance.

IC ICM C22F001-18  
ICS B21B021-00; B21C001-22; G21C003-06

CC 56-11 (Nonferrous Metals and Alloys)  
Section cross-reference(s): 71

ST zirconium alloy tube cracking resistance; stress  
corrosion cracking resistance tube; nuclear  
fuel cladding zirconium alloy tube

IT Pipes and Tubes  
(zirconium alloy for, stress-corrosion cracking-  
resistant, manuf. of)

IT Nuclear reactor fuels and fuel elements  
(claddings, zirconium alloys, stress-corrosion  
cracking-resistant, manuf. of)

IT Zirconium alloy, base  
RL: PEP (Physical, engineering or chemical process); PROC (Process)  
(tubes, with good stress-corrosion cracking  
resistance, manuf. of, for cladding of  
nuclear fuel)

IT 89342-04-1P  
RL: PEP (Physical, engineering or chemical process); PREP (Preparation);  
PROC (Process)  
(tubes, stress-corrosion cracking resistant

- , manuf. of, for cladding of nuclear fuel)
- IT 89342-04-1P  
RL: PEP (Physical, engineering or chemical process); PREP (Preparation);  
PROC (Process)  
(tubes, stress-corrosion cracking resistant  
, manuf. of, for cladding of nuclear fuel)
- L67 ANSWER 19 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
1992:536007 Document No. 117:136007 **Stress-corrosion cracking resistant zirconium alloy tubes for cladding of nuclear fuel.** Mae, Yoshiharu; Isobe, Takeshi (Mitsubishi Materials K. K., Japan). Jpn. Kokai Tokkyo Koho JP 04099254 A2 19920331 Heisei, 5 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1990-212640 19900810.
- AB The title **tubes** are **manufd.** by extruding **Zr alloy** to give raw **tubes**, recrystn. annealing of the raw **tubes** under vacuum for .gtoreq. 1 times, , Pilger rolling for .gtoreq.1 times, and stress-relief annealing at 430-490.degree. to give the o.d. redn. rate 1-30 %. Thus, an extruded **tube** of **Zr alloy** contg. Sn 1.5, Fe 0.2, and Cr 0.1% showed high **stress-corrosion cracking resistance.**
- IC ICM C22F001-18  
ICS B21B021-00; B21C001-22; C22C016-00; G21C003-06
- CC 56-11 (Nonferrous Metals and Alloys)  
Section cross-reference(s): 71
- ST zirconium alloy **tube** cracking resistance; **stress** corrosion cracking **resistance tube**; **nuclear fuel cladding** zirconium alloy **tube**
- IT **Pipes and Tubes**  
(zirconium alloy, with good **stress-corrosion cracking resistance, manuf. of**)
- IT **Nuclear reactor fuels and fuel elements**  
(**claddings, zirconium alloy tubes, stress** -corrosion cracking-**resistant, manuf. of**)
- IT Zirconium alloy, base  
RL: PEP (Physical, engineering or chemical process); PROC (Process)  
(**tubes, with good stress-corrosion cracking resistance, manuf. of, for cladding of nuclear fuel**)
- IT 89342-04-1P  
RL: PEP (Physical, engineering or chemical process); PREP (Preparation);  
PROC (Process)  
(**tubes, with good stress-corrosion cracking resistance, manuf. of, for cladding of nuclear fuel**)
- IT 89342-04-1P  
RL: PEP (Physical, engineering or chemical process); PREP (Preparation);  
PROC (Process)  
(**tubes, with good stress-corrosion cracking resistance, manuf. of, for cladding of nuclear fuel**)

L67 ANSWER 20 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
1992:138218 Document No. 116:138218 **Fatigue behavior of neutron irradiated Zircaloy-2 fuel cladding tubes.** Nakatsuka, Masafumi; Kubo, Toshio; Hayashi, Yo (Nippon Nucl. Fuel Dev. Co., Ltd., Oarai, Japan). ASTM Spec. Tech. Publ., 1132(Zirconium Nucl. Ind.), 230-45 (English) 1991. CODEN: ASTTA8. ISSN: 0066-0558.



AB The effects of n irradiation and I as a corrosive fission product on the fatigue behavior of **Zircaloy-2** fuel cladding tubes were investigated using 2 different types of test specimens to evaluate the **fatigue strength** of BWR fuel subjected to such variable loading conditions as load following or automatic frequency control operations. Fatigue life had a tendency to drop with increasing I partial pressure, reaching a saturation value approximately 1/10 of that in an inert gas atmosphere. Minimum I partial pressure affecting the fatigue behavior of fuel cladding tubes was estimated to be 0.1 Pa. This value was much higher than the calculated equilibrium vapor pressure of I in fuel rods, indicating that effects of I on the fatigue life would be very small or negligible during variable loading conditions. The n irradiation increased the fatigue life of cladding tube for the total strain amplitude >0.3% and decreased it <0.3%. The increase or decrease in fatigue cycles was attributed to the hardening effect or localized deformation in the irradiated material, respectively. Fatigue limit of unirradiated Zry-2 tubes was 0.22%, and n irradiation reduced the value to 0.18%. The total strain amplitude of 0.18% coincided with the elastic strain at the proportional limit under the uniaxial tensile test of irradiated Zry-2. Both for unirradiated and irradiated specimens, transgranular fracture surfaces were induced by the bending. Ductile fracture surfaces were observed for unirradiated material, and n irradiation changed this surface into a typical brittle transgranular one.

CC 71-5 (Nuclear Technology)

Section cross-reference(s): 56

ST neutron irradiation **Zircaloy** fuel cladding fatigue

IT **Pipes and Tubes**

(fatigue behavior of neutron-irradiated **Zircaloy-2** fuel-cladding)

IT **Nuclear reactor fuels and fuel elements**

(claddings, fatigue behavior of neutron irradiated **Zircaloy-2 tube BWR**)

IT 12586-31-1, Neutron

RL: PROC (Process)

(fatigue behavior of **Zircaloy-2** fuel cladding tubes bombarded by)

IT 7553-56-2, Iodine, properties

RL: PRP (Properties)

(fatigue behavior of **Zircaloy-2** fuel cladding tubes neutron irradiated in presence of)

IT **11068-94-3, Zircaloy-2**

RL: PROC (Process)

(fatigue behavior of neutron-irradiated fuel cladding tubes of)

IT 12586-31-1

RL: PROC (Process)

(nuclear reactor fuels and fuel elements, claddings, fatigue behavior of neutron irradiated **Zircaloy-2 tube BWR**)

IT **11068-94-3, Zircaloy-2**

RL: PROC (Process)

(fatigue behavior of neutron-irradiated fuel cladding tubes of)

L67 ANSWER 23 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1991:497493 Document No. 115:97493 **Manufacture** of zirconium alloy tubes. Harada, Makoto; Kanehara, Mitsuo; Abe, Katsuhiro (Kobe Steel, Ltd., Japan). Jpn. Kokai Tokkyo Koho JP 02270949 A2 19901106 Heisei, 4 pp. (Japanese). CODEN: JKXXAF. APPLICATION: JP 1989-93919 19890412.

- AB Molten **Zr alloy** is extruded, hot-rolled and cold-rolled to give a high-strength **tube** esp. useful for **nuclear fuel clad**. The **tube** shows corrosion crack **resistance** and high **creep** rupture **strength**.
- IC ICM C22F001-18  
ICS B21B023-00
- CC 56-11 (Nonferrous Metals and Alloys)  
Section cross-reference(s): 71
- ST zirconium alloy **tube** extrusion rolling
- IT **Pipes and Tubes**  
(zirconium alloy, extrusion and rolling of)
- IT Zirconium alloy, base  
RL: USES (Uses)  
(**tubes**, extrusion and rolling of, for **nuclear fuel clad tubes**)
- L67 ANSWER 26 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
1988:481675 Document No. 109:81675 Mechanical properties of zirconium alloy cladding **tubes** and critical fuel element power ramps. Novak, J.; Lauerova, D. (Inf. Cent., Nucl. Res. Inst., Rez, Czech.). Ustav Jad. Vyzk., [Rep.], UJV 8324 M, 12 pp. (English) 1988. CODEN: UJVYAK. ISSN: 0577-3857.
- AB Study of the mechanism of cladding dehermetization under power ramp conditions **made** it possible to substantiate and to refine an empirical correlation between hoop fracture strain measured after internal pressurization .epsilon.f and the crit. power ramp value .DELTA.NC. Relations of conventional mech. properties, of fracture **toughness** in 1 atm. KISCC and of .DELTA.NC were found. In a present version, a correlation of .epsilon.f and .DELTA.NC is based on a semiempirical model having certain phys. interpretation. To quantify this correlation, it is now sufficient to know a single point, based on power ramp expts. in a MTR. Typical .DELTA.NC values for BWR and PWR confirm the model predictions and thus may be considered as verification of the proposed model. A more precise assessment of .DELTA.NC = 13 kW/m was obtained for WWER fuel elements with Zr1Nb cladding, irradiated to satn. at 300.degree.. A task to find .DELTA.NC for higher cladding temps. typical for WWER-440 and WWER-1000 reactors is thus transformed to the detn. of the cladding mech. properties after irradiation at corresponding temps.
- CC 71-5 (Nuclear Technology)  
Section cross-reference(s): 56
- IT **Nuclear reactor fuels** and **fuel** elements  
(**claddings**, mech. properties of zirconium alloy, verification of model for)
- IT Zirconium alloy, base  
RL: PROC (Process)  
(mech. properties of cladding **tubes** of, verification of model for)
- IT 12742-60-8, **Zircaloy**  
RL: PROC (Process)  
(mech. properties of cladding **tubes** of, verification of model for)
- L67 ANSWER 28 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
1985:531028 Document No. 103:131028 Nuclear fuel element. Dodelier, Jacques; Melin, Philippe (Framatome et Cie., Fr.). S. African ZA 8406896 A 19850424, 10 pp. (English). CODEN: SFXXAB. APPLICATION: ZA 1984-6896 19840904. PRIORITY: FR 1983-14327 19830908.
- AB A nuclear fuel element was designed, which has greater reliability and simplification of **prodn**. The element comprises an open support

**tube** for the stack over at least a portion of its length, sepg. the cladding from the stack and sepd. from the cladding by a radial interval of initial thickness sufficient to retard the coming into contact of the **tube** and the cladding upon the swelling of the pellets under irradiation. The cladding in particular is **made up of a tube of Zr alloy** which has undergone a recrystallization treatment at 550-650.degree. so as to increase its **creep resistance** and **resistance** to the action of the coolant.

IC ICM G21C

CC 71-5 (Nuclear Technology)

ST reactor fuel cladding **tube**; zirconium alloy cladding **tube**

IT **Nuclear reactor fuels and fuel elements**  
(**claddings**, zirconium alloy open-support **tube** for)

IT Zirconium alloy, base

RL: PROC (Process)

(**nuclear reactor fuel cladding**  
**tube** from, with open support)

L67 ANSWER 29 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1985:208047 Document No. 102:208047 **Zircaloy tube** for **nuclear fuel cladding**. (Hitachi, Ltd., Japan).

Jpn. Kokai Tokkyo Koho JP 60026650 A2 19850209 Showa, 4 pp. (Japanese).  
CODEN: JKXXAF. APPLICATION: JP 1983-132712 19830722.

AB To **make** corrosion- and **stress** corrosion cracking-  
**resistant**, an extruded **Zircaloy 2 [11068-94-3**  
**] tube** is quenched from 870.degree., cold rolled slightly to give precise inner diam., annealed at 600.degree. for 2 h, pickled in aq. HNO3-HF, heated at 500.degree., inserted with a Zr **tube**, drawn in vacuum, electron beam-welded at both ends, oil press expanded to weld, cold rolled, annealed at 590.degree., cut, machined to remove the Zr liner, and acid pickled. The wt. gain is .apprx.100 mg/dm2 and nodular corrosion neg. in a conventional corrosion test, compared to .apprx.1000 mg/dm2 and pos. for the **tube** **prepd.** by a conventional process.

IC ICM C22F001-18

ICS B21C037-06; G21C003-20

CC 56-5 (Nonferrous Metals and Alloys)

Section cross-reference(s): 71

ST **Zircaloy tube nuclear fuel**  
**cladding**; thermomech treatment **Zircaloy tube**

IT **Pipes and Tubes**  
(**Zircaloy**, thermomech. treatment of, for **nuclear fuel cladding**)

IT **Nuclear reactor fuels and fuel elements**  
(**claddings**, thermomech. treatment of **Zircaloy tubes** for)

IT **11068-94-3**

RL: USES (Uses)

(thermomech. treatment of **tubes** of, for **nuclear fuel cladding**)

IT **11068-94-3**

RL: USES (Uses)

(thermomech. treatment of **tubes** of, for **nuclear fuel cladding**)

L67 ANSWER 30 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1985:35009 Document No. 102:35009 Cladding **tube** for reactor fuel element. (Genshi Nenryo Kogyo K. K., Japan). Jpn. Kokai Tokkyo Koho JP 59131196 A2 19840727 Showa, 5 pp. (Japanese). CODEN: JKXXAF.

APPLICATION: JP 1983-5186 19830118.

AB In a **cladding tube** for **nuclear fuel** elements based on an **alloy** (e.g. **Zr alloy**) having a hcp. structure, the **tube** is obtained in such a manner that the central crystal axis of the hcp. alloy structures **makes** an angle of 0.degree., .+-.30.degree., and .+-.30.degree. with respect to the radius of the **tube** at the inner surface of the **tube**, at the interior of the **tube**, and at the exterior surface of the **tube**, resp. The **tube** is more **resistant** to **stress** corrosion cracking, hydriding and peripheral direction stretching.

IC G21C003-06; B21B019-10

ICA C22C016-00

CC 71-5 (Nuclear Technology)

Section cross-reference(s): 56

ST fuel element cladding **tube** alloy

IT **Nuclear** reactor **fuels** and **fuel** elements  
(**claddings**, zirconium alloy **tubes**)

IT Containers

(**tubes**, zirconium alloy, for nuclear reactor fuel elements)

IT Zirconium alloy, base

RL: PROC (Process)

(**cladding tubes** of, for **nuclear**  
**fuel** elements)

L67 ANSWER 32 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1983:97690 Document No. 98:97690 Fuel rods for nuclear reactors. (Toshiba Corp., Japan). Jpn. Tokkyo Koho JP 57042199 B4 19820907 Showa, 4 pp. (Japanese). CODEN: JAXXAD. APPLICATION: JP 1978-162490 19781228.

AB The cladding **tube** for **prepg.** reactor fuel rod is obtained by depositing a Cu layer on the inner wall of a **Zr alloy tube**, oxidizing the Cu layer, then reducing the oxidized Cu layer. The pellet-cladding interaction is reduced and I-absorption capability is increased. The rod is useful in a LWR and is **resistant** towards **stress** corrosion cracking.

IC G21C003-06

CC 71-5 (Nuclear Technology)

ST copper coating **Zircaloy** cladding fuel; LWR fuel cladding copper layer; reactor fuel cladding copper layer

IT **Nuclear** reactor **fuels** and **fuel** elements  
(**claddings**, copper coated zirconium alloy, for reduced stress corrosion cracking and pellet-cladding interaction)

IT Zirconium alloy, base

RL: PROC (Process)

(**nuclear** reactor **fuel** element **cladding**  
**tube** of, copper coating for reduced stress corrosion cracking  
and fuel pellet-cladding interaction)

IT 7440-50-8, uses and miscellaneous

RL: USES (Uses)

(**nuclear** reactor **fuel** elements **cladding**  
**tube** of zirconium alloy coated by, for reduced stress corrosion  
cracking and pellet-cladding interaction)

L67 ANSWER 33 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1983:61692 Document No. 98:61692 Iodine-induced stress corrosion cracking of copper-barrier **Zircaloy-4 tubes**. Huang, Jenn Hwa; Lee, Tien; Chuang, Yii Der (Inst. Nucl. Energy Res., Lung-Tan, Taiwan). Ts'ai Liao K'o Hsueh, 13(1), 6-12 (English) 1981. CODEN: TLKHAJ. ISSN: 0379-6906.

AB The effect is discussed of vacuum-evapd. and electroplated thin Cu layers

on the stress corrosion susceptibility of barrier-type **Zircaloy-4** [11068-95-4] **tubes**. Pressurization tests were run on specimens from a single batch of **Zircaloy tubing** with various thickness of Cu layers at .apprx.573 and .apprx.633 K, resp. Specimens were internally pressurized with Ar contg. a nominal I concn. of 5 mg/cm2 of **Zircaloy** surface. The times-to-failure of the Cu-coated specimens were markedly longer as compared to those of the uncoated ref. specimens. No crack was obsd. on Cu films at **stresses** below the burst **strength** of the **tubes**. The Cu film reacted with I after extensive exposure in I vapor, and a brittle **product** was **formed** which might reduce the protectiveness of this plated layer. Though cleaner than the electroplated Cu film, the vacuum evapd. film was less compatible with **Zircaloy tubes** when its thickness exceeded a few microns. The higher purity of the vacuum-evapd. film did not benefit very much its stress corrosion cracking protectiveness.

CC 71-5 (Nuclear Technology)

Section cross-reference(s): 56

ST iodine stress corrosion cracking **Zircaloy**; fuel cladding **Zircaloy** iodine cracking; copper barrier **Zircaloy** iodine cracking; reactor fuel cladding copper barrier

IT **Pipes and Tubes**

(iodine-induced stress corrosion cracking of copper-barrier **Zircaloy-4**)

IT **Nuclear reactor fuels and fuel elements**

(**claddings**, iodine-induced stress corrosion cracking of copper-barrier **Zircaloy-4**)

IT 7440-50-8, reactions

RL: RCT (Reactant)

(iodine-induced stress corrosion cracking of **Zircaloy tubes** with barrier layer of)

IT **11068-95-4**

RL: PROC (Process)

(iodine-induced stress corrosion cracking of copper-barrier **tubes** of)

IT 7553-56-2, reactions

RL: RCT (Reactant)

(stress corrosion cracking of copper-barrier **Zircaloy tubes** induced by)

IT **11068-95-4**

RL: PROC (Process)

(iodine-induced stress corrosion cracking of copper-barrier **tubes** of)

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1982:604566 Document No. 97:204566 Beta-quenching of **Zircaloy** cladding **tubes** in intermediate or final size - methods to improve corrosion and mechanical properties. Andersson, T.; Vesterlund, G. (Sandvik AB, Sandviken, Swed.). ASTM Spec. Tech. Publ., 754(Zirconium Nucl. Ind.), 75-95 (English) 1982. CODEN: ASTTA8. ISSN: 0066-0558.

AB Three batches of **Zircaloy-2** [11068-94-3]

**tubing** were .beta.-quenched prior to the final cold-rolling, cold rolled 80 %, and annealed at 475 - 575.degree.. A 4th batch was .beta.-quenched in the final size. For comparison, std. **tubing** was included in all tests performed. The 2nd-phase particles were studied by means of optical and SEM. Corrosion testing was carried out at 400.degree. and in high-temp. (475 - 500.degree.) high-pressure steam. The mech. tests comprised tension, burst, and creep testing under internal pressure. .beta.-Quenching instead of an intermediate or the final anneal results in significant structural changes. The most striking features are

the **formation** of a structure consisting of plates of .alpha.-phase and the repptn. of much finer 2nd-phase particles in the plate boundaries. Cold-rolling of .beta.-quenched hollows followed by a final anneal in the .alpha.-range will give an equiaxed structure, but the size and distribution of the 2nd phase obtained in .beta.-quenching will not be markedly changed. The wt. gain at 400.degree. increases slightly as a result of .beta.-quenching in intermediate or final size. In high-pressure steam at 475 - 500.degree., on the other hand, such .beta.-quenching has a dramatic beneficial effect on the corrosion resistance. The short-term strength as measured in tension and burst testing is improved by .beta.-quenching of hollows or finished **tubes**, whereas such treatment results in a slight drop in ductility, esp. for **tubing** .beta.-quenched in the final size. The 400.degree. transverse **creep strength** is increased by the introduction of .beta.-quenching prior to the final cold-rolling. The improvement is caused mainly by small 2nd-phase particles, **formed** during .beta.-quenching, which gives rise to pptn. hardening.

CC 71-5 (Nuclear Technology)

ST **Zircaloy** cladding beta quenching; reactor fuel cladding corrosion prevention

IT **Nuclear** reactor **fuels** and **fuel** elements

(**claddings**, **Zircaloy**, .beta.-quenching of, for improved corrosion prevention and mech. properties)

IT **11068-94-3**

RL: PROC (Process)

(.beta.-quenching of cladding **tubes** of, for improved corrosion prevention and mech. properties)

IT **11068-94-3**

RL: PROC (Process)

(.beta.-quenching of cladding **tubes** of, for improved corrosion prevention and mech. properties)

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1981:111072 Document No. 94:111072 Behavior of unirradiated zirconium-lined and copper-plated **Zircaloy-2 tubing** under simulated PCI conditions. Gangloff, R. P. (Corp. Res. Dev. Dep., Gen. Electr. Co., Schenectady, NY, USA). Report, GEAP-25093, 109 pp. Avail. INIS; NTIS From: INIS Atomindex 1980, 11(23), Abstr. No. 565953 (English) 1979.

AB The expanding mandrel technique was used to evaluate the **fracture resistance** of unirradiated, cold-worked Zr and Cu barrier fuel cladding for simulated pellet-cladding interaction (PCI) conditions. A range of environment and loading conditions, shown to **produce** severe embrittlement of cold-worked **Zircaloy-2** [ **11068-94-3**] and including 4 Pa I2, pure Cd and molten Cd-satd. Cs, was employed for barrier screening.

CC 71-6 (Nuclear Technology)

Section cross-reference(s): 56

ST reactor fuel cladding embrittlement **Zircaloy**; BWR fuel cladding embrittlement **Zircaloy**; **fracture resistance** zirconium

IT **Nuclear** reactor **fuels** and **fuel** elements

(**claddings**, embrittlement of **Zircaloy**, fuel-cladding interactions in relation to)

IT 7440-43-9, uses and miscellaneous

RL: USES (Uses)

(as barrier screening for **Zircaloy** reactor fuel claddings)

IT 7440-46-2, uses and miscellaneous

RL: USES (Uses)

(barrier screening from cadmium-satd., for **Zircaloy** fuel

- claddings, embrittlement in relation to)
- IT 7553-56-2, uses and miscellaneous  
RL: USES (Uses)  
(embrittlement of **Zircaloy** exposed to fission-product  
)
- IT **11068-94-3**  
RL: PROC (Process)  
(embrittlement of reactor fuel claddings of)
- IT 7440-67-7, properties  
RL: PRP (Properties)  
(**fracture resistance** of unirradiated cold-worked)
- IT **11068-94-3**  
RL: PROC (Process)  
(embrittlement of reactor fuel claddings of)
- L67 ANSWER 39 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
1977:162543 Document No. 86:162543 Design of an irradiation device for the  
determination of the in-pile creep behavior of **Zircaloy** cladding  
**tubes** under internal and external overpressure, in FRG-2. Ahlf,  
J.; Reymann, A.; Eichhorn, O.; Gaertner, M. (Ges. Kernenergieverwert.  
Schiffbau Schiffahrt m.b.H., Geesthacht, Ger.). J. Nucl. Mater., 65(1),  
302-6 (English) 1977. CODEN: JNUMAM.
- AB The dimensional stability of fuel rods in light-water reactors is  
influenced by the **creep strength** of the  
**Zircaloy** cladding. Irradn. expts. were **made** in FRG-2 to  
det. the effect of n irradn. on the creep behavior of **Zircaloy**  
cladding. In these capsule expts. specimens can be tested in a He  
environment at temps. of 280-400.degree. in a fast n flux of .apprx.5  
.times. 10<sup>13</sup>/cm<sup>2</sup>-s under biaxial tensile and compressive stresses of  
70-150 N/mm<sup>2</sup>. The test equipment, the exptl. techniques, and the initial  
results are described.
- CC 71-5 (Nuclear Technology)  
Section cross-reference(s): 56
- ST app creep **Zircaloy** cladding
- IT **Nuclear reactor fuels and fuel elements**  
(**claddings**, creep behavior of **Zircaloy**, irradn.  
device for detn. of in-pile)
- IT **11068-95-4** 12742-60-8  
RL: PROC (Process)  
(cladding **tubes**, in-pile creep behavior of, irradn. device  
for detn. of)
- IT 12586-31-1, chemical and physical effects  
RL: PEP (Physical, engineering or chemical process); PROC (Process)  
(on creep behavior of **Zircaloy** cladding **tubes**,  
irradn. device for detn. of in-pile)
- IT **11068-95-4**  
RL: PROC (Process)  
(cladding **tubes**, in-pile creep behavior of, irradn. device  
for detn. of)
- L67 ANSWER 40 OF 42 HCAPLUS COPYRIGHT 2002 ACS  
1976:156987 Document No. 84:156987 Creep anisotropy of **Zircaloy**  
cladding **tubes**. Stehle, H.; Steinberg, E. (Kraftwerk Union  
A.-G., Erlangen, Ger.). Report, AED-CONF-75-170-004, 12 pp. Avail. INIS  
From: Nucl. Sci. Abstr. 1976, 33(4), Abstr. No. 7808 (German) 1975.
- AB Survey is given on the texture of **Zircaloy** [12742-60-8] cladding  
**tubes** obtained depending on the **manufacturing**  
conditions, and the state of knowledge on the anisotropy of the mech.  
properties of the zirconium alloys connected with the texture is outlined.  
Theor. formulations are set up for the phenomenol. representation of the

anisotropic creep. The results of tension and compression tests and the thus obtained creep site curves exhibit distinct differences with **tubes** having different textures. Furthermore, an asymmetry regarding compressive tensile stress is found in such a manner that the material under compression **stress** is more **resistant** to **creep**. Discussions follow on the deformation mechanisms, and a comparison with flow processes as well as indications on the significance of these creep results within the framework of fuel rod design are given.

CC 71-5 (Nuclear Technology)

Section cross-reference(s): 56

ST **Zircaloy** cladding **tube** creep; fuel cladding  
**Zircaloy**

IT **Nuclear reactor fuels** and **fuel** elements  
(cladding **tubes** for, creep anisotropy of  
**Zircaloy**)

IT **Pipes and Tubes**  
(creep anisotropy of **Zircaloy** cladding)

IT 12742-60-8

RL: PROC (Process)

(creep anisotropy of **cladding tubes** of, for  
**nuclear reactor fuels**)

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1971:483059 Document No. 75:83059 **Fabrication** technology and quality for **Zircaloy** fuel-cladding **tubes**. Yamamoto, Haruo; Okada, Takeshi; Nagai, Nobuyuki; Tanaka, Yoshiro (Kobe Steel, Ltd., Kobe, Japan). Karyoku Hatsuden, 21, 217-22 From: Nucl. Sci. Abstr. 1971, 25(10), 22041 (Japanese) 1970.

AB The quality of domestic **Zircaloy** fuel-cladding **tubes** was improved. The features of the established domestic technol. and quality control are described. The requirements for cladding **tubes**, in general, and the **production** procedure for **Zircaloy tubes** are explained. The **fabrication** of the **Zircaloy** cladding **tubes** are accomplished by cold-working. To improve the quality, it was necessary to have corrosion-produced Zr hydride in the circumferential direction. The 3 techniques, draw bench, pilger roll, and 3-roller roll, and the corresponding hydride behaviors, are described. The 3-roller rolling was best for the hydride problem and resulted in superior surface, dimensional accuracy, strength, and **toughness** properties.

CC 76 (Nuclear Technology)

ST **fabrication Zircaloy** fuel cladding **tubes**;  
rolling **Zircaloy** fuel cladding **tubes**

IT **Nuclear reactor fuels**, uses and miscellaneous  
(claddings, **Zircaloy**)

IT **Zircaloy**  
(**nuclear reactor fuel claddings**)

L67 ANSWER 42 OF 42 HCAPLUS COPYRIGHT 2002 ACS

1971:483057 Document No. 75:83057 **Fabrication** techniques and quality of **Zircaloy** cladding **tubes**. Yamamoto, Haruo; Okada, Takeshi; Onishi, Tadatoshi; Tanaka, Yoshiro (Kobe Steel, Ltd., Kobe, Japan). Kobe Seiko Giho, 20(2), 12-20 From: Nucl. Sci. Abstr. 1971, 25(10), 22039 (Japanese) 1970.

AB The **Zircaloy** cladding **tubes produced** by Kobe Steel Ltd. are superior in quality compared with those of other countries. The requirements for **Zircaloy** cladding **tubes** and the procedure adopted by Kobe Steel Ltd. are described, and various properties of the **produced tubes** are given. The requirements included corrosion resistance, compatibility with contained fuel, and



strength against the internal pressure of fission **product** gas.  
The overall procedure consisted of hot extrusion to give high purity and uniformity, cold processing to provide accurate dimensions, good **toughness**, and finally inspection. A 3-roller method used in the final process gave the highest difference between the redn. ratios of wall-thickness and outerdiam.

CC 76 (Nuclear Technology)

ST **fabricating Zircaloy** cladding fuel **tubes**;  
rolling **Zircaloy** cladding fuel **tubes**; extrusion  
**Zircaloy** cladding fuel **tubes**

IT **Nuclear** reactor **fuels**, uses and miscellaneous  
(claddings, **Zircaloy**)

IT **Zircaloy**  
(nuclear reactor fuel claddings)